FENOC
FirstEnergy Nuclear Operating Company

Perry Nuclear Power Plant
10 Center Road
Perry, Ohio 44081

William R. Kanda
Vice President - Nuclear

440-280-5579
Fax: 440-280-8029

September 8, 2003
PY-CEI/NRR-2735L

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Perry Nuclear Power Plant
Docket No. 50-440

Subject: Request for Enforcement Discretion Regarding Technical Specification (TS) 3.7.1, Emergency Service Water (ESW) System - Division 1 and 2, and TS 3.8.1, AC Sources-Operating

Ladies and Gentlemen:

This letter documents a request for regional enforcement discretion from compliance with the requirements of Perry Nuclear Power Plant (PNPP) Technical Specification (TS) 3.7.1, Emergency Service Water (ESW) System - Division 1 and 2, Required Actions B.1 and B.2; and TS 3.8.1, AC Sources – Operating, Required Actions F.1 and F.2.

The plant is operating at approximately 100% power. The "A" ESW system and divisional diesel generator were declared inoperable at 1717 hours eastern daylight time (EDT) on September 1, 2003, when low discharge pressure and flow indications were observed with the ESW "A" system. Disassembly of the ESW "A" pump indicated a shaft coupling sleeve failure. The apparent root cause of the ESW "A" pump failure is stress corrosion cracking of the 416 stainless coupling sleeve.

Due to the extent of the repair and the need to obtain replacement parts, the repair and post maintenance testing was not able to be completed within the specified TS completion time. Without enforcement discretion, the Limiting Condition for Operation (LCO) would have expired at 1717 on September 4, 2003, requiring the plant to be in at least MODE 3 by 0517 hours EDT on September 5, 2003. The PNPP requested that the Nuclear Regulatory Commission exercise discretion not to enforce compliance with the required completion time for TS 3.7.1, Required Actions B.1 and B.2; and TS 3.8.1, Required Actions F.1 and F.2 for a 72 hour period, to avoid a plant shutdown that would impose an unnecessary plant transient without a significant offsetting safety benefit.

Attachment 2 provides the information specified in NRC Inspection Manual, Part 9900: Technical Guidance, Operations – Notices of Enforcement Discretion," dated November 2, 2001. This attachment provides the bases and justification that the public health and safety is maintained.

This enforcement discretion request was verbally transmitted to members of the NRC staff during a teleconference on September 4, 2003. The NRC granted the Notice of Enforcement Discretion on September 4, 2003, contingent upon review of changes by the Plant Operations Review Committee and approval by the Plant Manager. This contingency

A001

was successfully satisfied and communicated to Mr. Mark Ring, NRC Region III, during a phone call at 1645 EDT, September 4, 2003.

Please refer questions regarding this submittal to Mr. Vernon Higaki, Manager – Regulatory Affairs, at (440) 280-5294.

Sincerely

Attachments:

1. Notarized Affidavit
2. Request for Enforcement Discretion

cc: NRC Region III Administrator
NRC Resident Inspector
NRC Project Manager

I, William R. Kanda, hereby affirm that (1) I am Vice President - Perry, of the FirstEnergy Nuclear Operating Company, (2) I am duly authorized to execute and file this certification as the duly authorized agent for The Cleveland Electric Illuminating Company, Toledo Edison Company, Ohio Edison Company, and Pennsylvania Power Company, and (3) the statements set forth herein are true and correct to the best of my knowledge, information and belief.

William R. Kanda

Subscribed to and affirmed before me, the 8th day of September, 2003

Jane E Mott

JANE E. MOTT
Notary Public, State of Ohio
My Commission Expires Feb. 20, 2005
(Recorded in Lake County)

REQUEST FOR ENFORCEMENT DISCRETION

1. The Technical Specification or other license conditions that will be violated.

Regional enforcement discretion was requested and granted from compliance with Technical Specifications (TS) Limiting Condition for Operation (LCO) 3.7.1, Required Actions B.1 and B.2; and LCO 3.8.1, Required Actions F.1 and F.2. Specifically, this one time request is for an additional 72 hours to accomplish restoration of the Division 1 Emergency Service Water (ESW) "A" subsystem to operable status.

Perry Nuclear Power Plant (PNPP) TS LCO 3.7.1 requires the Division 1 and 2 ESW subsystems to be operable in MODES 1, 2, and 3. With one Division 1 or Division 2 ESW subsystem inoperable, Required Action A.1 requires restoration of the ESW subsystem to operable status with a Completion Time of 72 hours. If the Completion Time for this Required Action is not met, then the plant must be brought to MODE 3 within 12 hours and MODE 4 within 36 hours (Required Actions B.1 and B.2 respectively).

Also, in accordance with the Notes of LCO 3.7.1, if one Division 1 or Division 2 ESW subsystem is inoperable, the applicable Conditions and Required Actions of LCO 3.4.9, "Residual Heat Removal (RHR) Shutdown Cooling System-Hot Shutdown," and LCO 3.8.1, "AC Sources-Operating," are required to be entered. LCO 3.4.9 for RHR Shutdown Cooling System-Hot Shutdown is only applicable in MODE 3 and therefore, did not require any action in this instance.

LCO 3.8.1 is applicable in MODES 1, 2 or 3 for the Division 1 Emergency Diesel Generator, with a Completion Time for Required Action B.4 of 14 days. This Completion Time would not be challenged by the 72 hour extension being sought by this request for enforcement discretion. However, the TS Bases for Required Action B.4 contain guidance which implies that use of the period between 3 days and 14 days (≥72 hours) requires risk-significant systems that depend on the Division 1 Emergency Diesel Generator be operable. ESW inoperability results in a number of other Division 1 systems being declared inoperable. Based on the implication in the current Bases words, the approved enforcement discretion request included LCO 3.8.1 Condition F, which would require the plant be brought to MODE 3 within 12 hours and MODE 4 within 36 hours (Required Actions F.1 and F.2 respectively).

As a result of the inoperability of the ESW "A" system, the following are considered inoperable:

- Low Pressure Core Spray system
- Reactor Core Isolation Cooling system
- Division 1 Emergency Diesel Generator
- Residual Heat Removal "A" system
  Modes:
  - Low Pressure Coolant Injection
  - Suppression Pool Cooling
  - Shutdown Cooling
  - Containment Spray
- Control Room HVAC Train "A"
- Hydrogen Analyzer "A"

- Emergency Closed Cooling system "A"
- Combustible Gas Mixing Compressor "A"

With the exception of the Division 1 Emergency Diesel Generator, the Conditions and Required Actions related to the above inoperable systems are not entered since TS 3.0.6 provides an exception to LCO 3.0.2. TS 3.0.6 allows an exception to entering the Conditions and Required Actions by specifying that when a support system is inoperable (in this case ESW "A") and there is an LCO specified for it in the Technical Specifications, the supported systems are declared inoperable but it is not necessary to enter into the supported systems' Conditions and Required Actions unless directed to do so by the support system's Required Actions.

2. The circumstances surrounding the situation, including apparent root causes, the need for prompt action and identification of any relevant historical events.

At 1635, on Monday, September 1, 2003, the Emergency Service Water (ESW) system "A" pump was started to provide dilution flow to facilitate draining a leaking sodium hypochlorite line. At 1707, the draining evolution was completed and no further leakage was observed.

At 1717, the control room received the following annunciators: "ESW FROM ECC HX A FLOW LOW", "ESW PUMP A DISCHARGE PRESSURE LOW", "ESW TO RHR A HX'S FLOW LOW", and " ESW TO DIESEL HEAT EXCHANGER FLOW LOW." All of these alarms are associated with the Division 1 ESW subsystem.

Control Room panel indications for the ESW Pump "A" subsystem were motor amps at approximately 20 amps and pump discharge pressure at approximately 20 psig. All indications for ESW flow to Residual Heat Removal (RHR) "A" heat exchanger, Emergency Closed Cooling (ECC) "A" heat exchanger, and Division 1 Emergency Diesel Generator heat exchanger were 0 gpm.

A Perry Plant Operator (PPO) was dispatched to the ESW pump house. At 1728, he reported that the ESW pump "A" was rotating and the local discharge pressure indication was 20 psig. The ESW Pump "A" was stopped and all ESW "A" low flow alarms reset as expected.

The Division 1 Emergency Diesel Generator and ESW "A" system were then placed in secured status at 1731 and 1757 respectively.

A diver inspection on Tuesday, September 2, at 0300 hours revealed the first stage impeller was not rotating when the ESW "A" pump motor was rotated by hand. A complete pump disassembly was then required.

<u>Apparent Root Cause</u>

The apparent root cause of the ESW "A" pump failure is stress corrosion cracking of the 416 stainless coupling sleeve. A combination of factors including the 416 stainless coupling material sensitivity to corrosion, high sulfur content and low ductility, run time of the pump, the lake water environment which includes chlorides, and additional stress caused by improper field installation of the pump coupling sleeve caused the premature

failure of the pump. It should be noted that although sodium hypochlorite is injected into the ESW basins, it dilutes readily and is not anticipated to have caused this failure.

A full disassembly of the ESW "A" pump revealed that the upper coupling sleeve on the pump shaft had failed. The sleeve was cracked into two pieces. The crack was longitudinal and originated at one corner of the machined slot for the key (keyway) and propagated to the opposite corner at the other end. The bottom of the keyway is machined 90 degrees with no radius, and thus provides a stress riser location.

The cracked coupling pieces were removed and inspected. All failed components were retrieved from within the pump assembly. During inspection it was determined that the coupling sleeve was not properly centered over the slip rings when it was installed on the pump shaft. The coupling sleeve has two locking screws; one on each side of the coupling located 180 degrees apart. The two locking screws are intended to be bottomed out on the coupling sleeve surface and be screwed into a notch provided in the split ring collar. It can be seen on the cracked coupling sleeve that the locking screw was not installed in the recess of the split collar and the coupling sleeve was installed off center leaving approximately one inch (one quarter of the total length) of the key extending beyond the coupling sleeve. Review of the maintenance installation instruction utilized for pump reassembly concluded that this instruction was deficient in providing the required details to ensure consistent and proper coupling installation.

Retrieved parts from the original equipment manufacturer (OEM) failed pump coupling, including both sleeve and split rings, were forwarded to FirstEnergy's BETA Laboratory for failure analysis. A preliminary metallurgical failure analysis was performed, which included visual examinations, chemical analysis via vacuum spectrometry, hardness and metallography. No deficiencies in material composition were identified. However, the results of the metallurgical investigation indicate that stress corrosion cracking was involved in the failure mechanism.

For stress corrosion cracking to occur, the following three conditions need to be present concurrently: (1) a susceptible material, (2) tensile stress which naturally occurs during pump operation and (3) an environment. For the subject shaft coupling application, all three conditions exist to some degree. The tensile stress parameter, in particular, may have played a key role in the failure scenario.

That is, additional tensile stresses from the incorrect sleeve installation were induced not just by the locking screws bearing onto the split ring, but also by a substantial reduction in shear key bearing area. The increase in tensile stress within the coupling sleeve was probably substantial (estimated to be in excess of 30%); thereby, taking peak operating stresses in the sleeve close to/above material allowables.

The relatively brittle nature of the coupling material (Type 416) also helps to explain the coupling sleeve failure into two separate sections. After the crack initiated at the keyway and started to propagate, the coupling sleeve most probably deformed enough under load to allow the shear key to rotate (slip) relative to the sleeve itself; thus causing a different sleeve tensile loading mechanism. This different loading, in conjunction with the brittle nature of the Type 416 material, caused a second, separate failure plane to occur within the sleeve. A more ductile sleeve material may have undergone the deformation without a second failure plane occurring.

In conclusion, the apparent root cause of the coupling failure is time dependent involving the initiation of stress corrosion cracking of the coupling sleeve with the time factor accelerated by an increased stress condition induced by improper coupling installation.

Industry Experience

Review of PNPP operating history concluded that the coupling failure associated with the ESW "A" pump is the first occurrence of such a failure of this design at Perry.

Review of FENOC experience identified a similar coupling failure that occurred at the Beaver Valley Power Plant (BV) in early 2002 as documented in the BV corrective action program. The coupling failure at BV involved a threaded coupling made of 416 SS on a similar size OEM Pump with comparable environmental conditions and service history. The failure cause of the BV pump was also stress corrosion cracking on a 416 stainless coupling. The BV pump was in a non-safety related application. Perry personnel involved in the investigation of the ESW A pump coupling failure have been in contact with BV personnel involved in the BV coupling failure.

Preliminary discussions with the OEM indicate minimal experience with failures of 416 SS in raw water environments.

Additional Industry review is on-going.

Extent of Condition

An extent of condition review was performed that looked at a variety of safety and non-safety related pumps installed at PNPP for configuration similarities, coupling similarities, and written instruction similarities. The ESW "B" pump is essentially identical to the ESW "A" pump and the ESW "C" pump is also very similar. However, the extent of condition review identified no immediate operability concerns for either the "B" or "C" ESW pumps.

The ESW "B" pump was last rebuilt in April 2003 (Refuel Outage 9) using all new couplings, bearings, shafts, split rings, etc. with the replacement pump parts purchased from the OEM prior to the rebuild. Even though this pump was reassembled using the same maintenance instructions, interviews conducted with maintenance personnel provide reasonable assurance that the couplings were indeed installed correctly during the April 2003 pump rebuild. In its current condition, the ESW "B" pump has been in service for approximately 4 months with minimal run time. In comparison, the ESW "A" pump was in service for over 6 years with approximately 6000 hrs total run time prior to experiencing its coupling failure. As such, based on the existing pump having new parts installed, having minimal inservice/run time, and having reasonable assurance of proper coupling installation, the ESW "B" pump is not considered to have any immediate operability issues.

The existing ESW "C" pump went into service in 1985. The present configuration is the original installation during plant construction with the pump shaft/coupling components having never been re-assembled by plant personnel. Consequently, it is not likely that the couplings on this pump have been installed incorrectly because plant personnel have not used the installation criteria in the current maintenance installation instruction on this pump. The ESW "C" pump supports the High Pressure Core Spray system and associated diesel generator. As such, its annual run time is approximately one-tenth of

the time of the larger ESW pumps. The estimated total in service run time on the ESW "C" pump is approximately 2000 hours since initial start-up which is only about one-third of the run hours of the ESW "A" pump at the time that it experienced its failure. Also, preliminary review of the OEM seismic qualification report indicates that the applied shear stress on the ESW "C" pump's coupling is approximately 25% less than those in the ESW "A" and "B" pumps. As such, based on the limited inservice run time of the ESW "C" pump, lower operating coupling stresses, together with reasonable assurance that the couplings were installed correctly under OEM oversight during plant construction, the ESW "C" pump is not considered to have any immediate operability issues.

Corrective actions associated with this issue include follow up inspection of the ESW "B" and "C" pumps during the current operating cycle but no later than the next refueling outage.

3. **The safety basis for the request, including an evaluation of the safety significance and potential consequences of the proposed course of action. This evaluation should include at least a qualitative risk assessment using both risk insights and informed judgements, as appropriate.**

Safety Basis/Risk Impact

The Notice of Enforcement Discretion (NOED) request was evaluated from a probabilistic risk perspective. This evaluation concluded that there is less risk related to operating the plant with ESW "A" out of service for an additional 72 hours beyond the associated 72 hour Completion Time when compared to the risk associated with that of a plant shutdown.

The methodology applied in this evaluation evaluates the impact to Core Damage Probability (CDP) and Large Early Release Probability (LERP) as follows:

CDP

The PNPP Probabilistic Safety Assessment (PSA) models ESW "A" as a support system to several systems. When ESW "A" is failed, systems supported by ESW "A" will also be failed in the PSA model. The baseline CDF using the PNPP PSA model is 5.904E-06 per year. With all risk significant equipment except train "A" of ESW (and its supported systems) available the CDF is quantified to be 1.063E-05 per year. The difference in risk from the baseline configuration relative to the ESW "A" unavailable configuration is 4.726E-06 per year or 1.295E-08 per day. Over a 72-hour period the increase in risk expressed as a Incremental Conditional Core Damage Probability (ICCDP) would be 3.89E-08.

The risk incurred during operation was qualitatively compared to the risk of shutting the reactor down and performing repairs off-line. More challenges are typically experienced during plant startup and shutdown evolutions than during steady state operations. Therefore, the risk incurred during a shutdown or startup will be greater than that incurred during normal operation. The Conditional Core Damage Probability (CCDP) due to a reactor scram with the condenser available and ESW "A" unavailable can be estimated using the PSA model. Plant initiators in the PSA model are tabulated and modeled to represent the response of the plant within a power band between 20 percent

and full power. Setting the initiating event (reactor scram with condenser available) equal to one and all other initiating events to 1.0E-09 a CCDP of 3.212E-07 is obtained.

The risk associated with a controlled shutdown is different than the risk incurred during a scram. During a reactor scram the initial response of the plant is controlled by automatic actuation of equipment. Several of these automatic actions are replaced with operator actions during a controlled shutdown. Operator actions introduce the potential for operator errors and subsequent perturbations on plant systems. These perturbations would further challenge operations and plant systems.

Based on these results, it is believed that the ICCDP incurred during 72 hours of operations with train "A" of ESW unavailable (3.89E-08) would be less than the CCDP incurred during a reactor shutdown with ESW "A" unavailable.

LERP

The PNPP baseline Large Early Release Frequency (LERF), which includes internal flooding, is 2.08E-07 per year. Modifying the Level 2 model to address the configuration with all risk significant systems available except train "A" of ESW (and its supported systems), the LERF is computed to be 2.51E-07 per year. The increase in LERF incurred while operating with ESW "A" unavailable is 4.3E-08 per year or 1.2E-10 per day. Therefore, the increase in risk due to LERF while operating with ESW "A" unavailable is small.

Qualitatively the Large Early Release Probability (LERP) associated with a plant shutdown with the "A" train of ESW unavailable can be approximated by the LERP incurred during a reactor scram with the condenser and all other equipment except ESW "A" initially available. From a PSA perspective, a reactor scram from 20 percent power would result in a similar response of plant systems as a scram from full power. There could be some minor adjustments to operator performance for a controlled shutdown; however, the differences are not expected to significantly sway the results. Using the PSA model documented in Calculation PSA-020, Revision 0, the LERP due to a reactor scram with all equipment available except ESW train A is computed to be 4.83E-09. The LERP incurred during 72 hours of operation with ESW train "A" unavailable is 3.6E-10. This is less than the LERP that would be incurred during a reactor scram.

While the loss of offsite power and station blackout are the dominant contributors to core damage at Perry, the impact of a disruption in offsite power is deemed to be as severe during shutdown as it is at power.

Conclusion

The request for enforcement discretion has been assessed from a probabilistic risk perspective. This assessment concluded that there would be no net increase in the radiological risk due to continued operation of the plant while ESW pump "A" is repaired relative to shutting down and performing the repair off-line. This conclusion is based on results that indicate that the increase in the Incremental Conditional Core Damage Probability (ICCDP) incurred during 72 hours of operation with ESW pump "A" unavailable would be less than the CCDP incurred during a reactor scram. It is believed that both the CDP and LERP associated with operating for 72 hours with ESW train "A" unavailable are less than the CDP and LERP that would be incurred during a reactor

shutdown. There is a risk associated with a plant shutdown. At reduced power, operation of some equipment is transferred to manual control. During the transition, equipment not previously operating will be required to be placed in service. The possibility of equipment failures during the transition phase is more likely than at steady state operation. Manual manipulation of equipment during power maneuvers also introduces the potential for human errors that are not present during steady state operation.

Transient events are also more likely to occur during the transition phase. The frequency of events such as the loss of feedwater or the loss of offsite power is higher during power transitions. System perturbations brought on by shutdown maneuvers may challenge operators when their primary focus is on maintaining the functionality of systems required for shutdown. These potential problems would be compounded by the unavailability of ESW train "A".

Risks incurred during shutdown and again during startup have not been quantified. Risks incurred during low power transition, shutdown, and startup, would increase the total risk quantified for a forced outage. It is believed that continued operation is justified by only considering the risk that would be incurred during a reactor shutdown. It is also believed that the risk due to a shutdown can be approximated by a reactor scram using the PSA model.

Therefore, this assessment concludes that there is no net increase in risk by extending operation by 72 hours beyond the 72 hour Completion Time with ESW "A" unavailable, and that there is less risk impact of operating in this configuration than shutting down in this configuration.

4. The justification for the duration of the noncompliance.

The PNPP proposes to exceed the 72 hour action time, which expires at 1717 hours on September 4, 2003 by 72 hours to allow sufficient time to restore the ESW "A" subsystem to an operable status. This additional time is required to repair the ESW "A" pump by replacing all four of its shaft coupling assemblies. Additional time is necessary to obtain parts, complete the installation of the pump, and complete the required post maintenance/surveillance testing of the pump to restore it to operable.

The current schedule includes five hours each to install the four shaft couplings (twenty hours total), four hours to install the discharge head, five hours to assemble the stuffing box, six hours to install motor and pump auxiliaries, four hours to reterminate the motor, two hours for packing, and six hours to complete fill and venting activities, post maintenance testing and paperwork closure. This work is scheduled to be complete within 48 hours. The request for 72 hours allows for delivery of contingency parts.

Also, there are no surveillances on Division 2 or 3 equipment that will be due within the 72 hours requested which would require additional discretionary time.

As discussed above, there is no net increase in risk associated with operating the plant for an additional 72 hours.

5. **The basis for the licensee's conclusion that the noncompliance will not be of potential detriment to the public health and safety and that no significant hazard consideration is involved.**

   The PNPP staff has evaluated this request for enforcement discretion against the criteria set forth in 10 CFR 50.92 and concludes that the request involves no significant hazards consideration. The evaluation is provided below.

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

      The requested action does not physically alter any plant structures, systems, or components, and does not affect or create new accident initiators or precursors. The Required Action Completion Time is not an accident initiator; therefore, there is no effect on the probability of accidents previously evaluated.

      The ESW "A" subsystem is required to mitigate the consequences of accidents previously evaluated in the Updated Safety Analysis Report. The redundant train (ESW "B") remains operable and capable of performing its intended function. The requested extension does not affect the types or amounts of radionuclides released following an accident, or the initiation and duration of their release.

      Therefore, the probability of occurrence or the consequences of accidents previously identified are not significantly increased.

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

      Exceeding the 72 hour allowed outage time by an additional 72 hours does not introduce new failure modes or mechanisms associated with plant operation. Furthermore, the additional 72 hour period associated with the restoration of the ESW "A" subsystem would not create a new accident type.

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

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

      The applicable margin of safety is the period of time that the ESW "A" subsystem is inoperable. PNPP has determined that there is no net increase in risk associated with exceeding the 72-hour allowed outage time by an additional 72 hours.

      Although the proposed action deviates from a requirement in LCO 3.7.1, it does not affect any safety limits, other operational parameters, or setpoints in the TS, nor does it affect any margins assumed in the accident analysis. The ESW "B" and "C" subsystems continue to be operable and capable of performing their required design function.

6. The basis for the licensee's conclusion that the noncompliance will not involve adverse consequences to the environment.

Perry has evaluated the requested enforcement discretion request against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. Perry has determined that the requested action meets the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9). This determination is based on the fact that the proposed action is being requested as enforcement discretion to a license issued pursuant to 10 CFR 50, and that the change involves no significant hazards considerations. Although the proposed action involves noncompliance with the requirements of an LCO:

a) The proposed action involves no significant hazards consideration.

b) There is no significant change in the types or a significant increase in the amounts of any effluent that may be released off-site, since the proposed action does not affect the generation of any radioactive effluent nor does it affect any of the permitted release paths.

c) There is no significant increase in individual or cumulative occupational radiation exposure. The action proposed in this request for enforcement discretion will not significantly affect plant radiation levels, and, therefore does not significantly affect dose rates and occupational exposure.

Accordingly, the proposed action meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9).

7. Any proposed compensatory measures.

a) The following actions were taken when the ESW "A" system was declared inoperable:

1. The following areas were posted as protected in the plant and control room:
   Division 2 and 3 D/G
   Division 2 and 3 AC/DC electrical systems
   Emergency Service Water B and C
   Emergency Closed Cooling B
   Residual Heat Removal systems B and C
   High Pressure Core Spray
   Motor Feed Pump
   Diesel Fire Pump
   Bus L10
   RCIC Pump Room

2. All scheduled work activities were evaluated, and the schedule was appropriately changed for the current plant risk.

b) While in the discretionary period, it is expected that as a minimum the following compensatory actions will be implemented to minimize the risk that is within the control of the plant.

- Significantly risk sensitive areas will continue to be posted and access restricted.
- No discretionary maintenance or testing will be performed on risk significant equipment. This includes maintaining the offsite power system and the switchyard.
- Work on ESW "A" will be maintained around the clock and the Work Support Center will be staffed for the duration of the discretionary enforcement period.
- All Division 2 and 3 equipment will be maintained operable and reviewed as required by TS.

  The PNPP will anticipate the need for suppression pool cooling and will commence a reactor shutdown such that RHR "B" will not be planned to be placed in suppression pool cooling prior to Cold Shutdown.
- The High Pressure Core Spray system will be maintained operable.
- Load dispatcher has been notified to suspend work that could affect the stability of offsite power to the Perry switchyard.
- Operator briefs will be held prior to every shift to discuss the enforcement discretion, plant ESW status, and contingency requirements.

8. **A statement that the request has been approved by the facility organization that normally reviews safety issues (Plant On-site Review Committee, or its equivalent).**

   On September 4, 2003, this request was reviewed by the Plant Operations Review Committee (PORC) for the Perry Nuclear Power Plant, and recommended for approval by the Plant Manager. The Plant Manager concurred with the recommendation.

9. **The request must specifically address which of the NOED criteria for appropriate plant conditions specified in Section B is satisfied and how it is satisfied.**

   For the operating Perry Nuclear Power Plant, this enforcement discretion is intended to avoid undesirable transients as a result of forcing compliance with the license condition (TS 3.7.1, Required Actions B.1 and B.2; and 3.8.1, Required Actions F.1 and F.2), thus minimizing potential safety consequences and operational risks associated with a plant shutdown.

10. **If a follow-up license amendment is required, both the written NOED request and the license amendment request must be submitted within 2 working days. The licensee's amendment request must describe and justify the exigent circumstances (see 10 CFR 50.91(a)(6)).**

    A follow-up amendment is not required due to the short duration in which the enforcement discretion is to be effective.

11. **For severe weather or other natural phenomena-related NOEDs, the licensee's request must be sufficiently detailed for the staff to evaluate the likelihood that the event could affect the plant, the capabilities of the ultimate heat sink, on-site and off-site emergency preparedness status, access to and from the plant, acceptability of any increased radiological risk to the public and the overall public benefit.**

    This NOED does not involve severe weather thus this request does not contain related information.