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U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Perry Nuclear Power Plant
Docket No. 50-440
Technical Specification Change
Request: Miscellaneous Technical
and Administrative Changes

Gentlemen:

Enclosed is a request for amendment to the Perry Nuclear Power Plant (PNPP) Unit 1 Facility Operating License NPF-58. In accordance with the requirements of 10CFR50.91(b)(1), a copy of this Amendment Request has been sent to the State of Ohio as indicated below.

This Amendment Request proposes revision of PNPP Technical Specifications to incorporate miscellaneous technical and administrative changes determined not likely to involve significant hazards considerations in accordance with previously published Commission guidance (51 FR 7751, March 6, 1986). Attachment 1 provides a summary of the proposed changes. Attachment 2 provides a copy of the marked up Technical Specification pages.

If you have any questions, please feel free to call.

Sincerely,

M. D. Lyster

Michael D. Lyster

MDL:CJF:ss

Attachments

cc: NRC Project Manager
NRC Resident Inspector Office
NRC Region III
State of Ohio

A001
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I. SUMMARY/SAFETY ANALYSIS

This Amendment Request proposes numerous technical and administrative changes determined not likely to involve significant hazards considerations in accordance with previously published Commission guidance (51 FR 7751, March 6, 1986). A Summary/Safety Analysis of each proposed change is provided below:

(1) TECHNICAL SPECIFICATION(S): Specification 3.3.1, "Reactor Protection System Instrumentation" ACTION a; Specification 3.3.2, "Isolation Actuation Instrumentation" ACTION b; and Specification 3.3.3, "Emergency Core Cooling System Actuation Instrumentation" Table 3.3.3-1, ACTION 38.

PAGE NUMBER(S): 3/4 3-1, 3/4 3-9 and 3/4 3-31

DESCRIPTION OF PROPOSED CHANGE(S): (1) Delete the last sentence of Specification 3.3.1 ACTION a, and of Specification 3.3.2 ACTION b, which state that the provisions of Specification 3.0.4 are not applicable. Refer to Attachment 2, page 1 and 2 of 18, for a copy of marked up Technical Specification pages 3/4 3-1 and 3/4 3-9. (2) Add a footnote to Specification 3.3.3, Table 3.3.3-1 ACTION 38, which states that the provisions of Specification 3.0.4 are not applicable. Refer to Attachment 2, page 3 of 18, for a marked up copy of Technical Specification page 3/4 3-31.

JUSTIFICATION FOR PROPOSED CHANGE(S): Amendment 30 to PNPP's Unit 1 Facility Operating License, issued May 24, 1990, implemented several changes to PNPP Technical Specifications based on previously published Commission guidance contained in Generic Letter 87-09, dated June 4, 1987, as an initiative to improve standard Technical Specifications. The Generic Letter 87-09 changes were requested for the Perry Plant by letter PY-CEI/NRR-0720L dated September 17, 1987 with additional supporting documentation provided to the NRC staff by letters PY-CEI/NRR-1122L dated March 12, 1990, and PY-CEI/NRR-1183L dated June 8, 1990.

Among the changes to PNPP Technical Specifications implemented by Amendment 30, based upon the Generic Letter 87-09 guidance, was a change to Specification 3.0.4. The Generic Letter 87-09 modification to Specification 3.0.4 removed unnecessary restrictions on changes in operational modes or other operating conditions when the ACTION requirements defined remedial measures that permitted unlimited continued operation. Prior to the Amendment 30 change to Specification 3.0.4, mode changes could only be made when the plant was being operated under the provisions of ACTION requirements if a specific exception to the requirements of Specification 3.0.4 was provided within individual Specifications.

As a consequence of the Generic Letter 87-09 modification to Specification 3.0.4, individual Specifications with Action Requirements permitting continued operation for an unlimited period of time no longer needed to indicate that Specification 3.0.4 does not apply. Consequently, Amendment 30 also revised numerous

individual specifications to delete the noted exception to avoid confusion about the applicability of Specification 3.0.4.

Individual Specification 3.3.1 ACTION a, and Specification 3.3.2 ACTION b, still contain such an exception indicating that Specification 3.0.4 does not apply. As a consequence of the Generic Letter 87-09 modification to Specification 3.0.4, these exceptions to Specification 3.0.4 are no longer needed and should have been included among the set of Specification 3.0.4 exceptions deleted by Amendment 30. Deletion of the Specification 3.0.4 exception provisions to Specification 3.3.1 ACTION a, and Specification 3.3.2 ACTION b, is consistent with the guidelines provided by the NRC staff in Generic Letter 87-09 since conformance to these Actions (placing the inoperable channels in the tripped condition) permits continued operation for an unlimited period of time. This proposed change will help to avoid confusion about the applicability of modified Specification 3.0.4.

Unlike the above two Specifications for which the 3.0.4 exception should have been deleted, Specification 3.3.3 Table 3.3.3-1 ACTION 38 contained a Specification 3.0.4 exception which was deleted by Amendment 30 but should not have been. Prior to Amendment 30, Table 3.3.3-1 ACTION 38 contained a Specification 3.0.4 exception. Unlike the two examples discussed above, however, Table 3.3.3-1 ACTION 38 does not permit continued operation for an unlimited period of time once the specified remedial action is taken. Rather, Table 3.3.3-1 ACTION 38 requires placing the inoperable channel in the tripped condition, then allows operation to continue only "until performance of the next required CHANNEL FUNCTIONAL TEST." At that time, an acceptable CHANNEL FUNCTION TEST would satisfy the LCO and thereby allow normal operation to continue. An unacceptable CHANNEL FUNCTIONAL TEST would require plant shutdown.

The purpose of the original Specification 3.0.4 exception to Table 3.3.3-1 ACTION 38 was to provide the flexibility to allow entry into higher modes of operation during this allowable period of continued operation. However, during preparation of the License Amendment Request that implemented Generic Letter 87-09, Table 3.3.3-1 ACTION 38 was inadvertently included among the set of individual Specifications with Action Requirements permitting unlimited continued operation (which therefore no longer needed to indicate that Specification 3.0.4 does not apply). Once included in this category, Table 3.3.3-1 ACTION 38 was revised by Amendment 30 to delete the noted exception.

In accordance with the NRC staff's recommendations contained in Generic Letter 87-09, page 3, paragraph 2, exceptions to Specification 3.0.4 should not be deleted for individual specifications if a mode change would be precluded by Specification 3.0.4 as revised. The Specification 3.0.4 exception contained in Table 3.3.3-1 ACTION 38 is one such Specification 3.0.4 exception which should not have been deleted because Table 3.3.3-1 ACTION 38 would not satisfy the provisions under which mode changes are

permitted by the Amendment 30 revision to Specification 3.0.4. The requested change to Table 3.3.3-1 ACTION 38, to reinsert the Specification 3.0.4 exception previously deleted by Amendment 30, will be consistent with the NRC staff's recommendations contained in Generic Letter 87-09, wherein it was stated that it is not the staff's intent that the revision of Specification 3.0.4 (pursuant to the recommendations contained in Generic Letter 87-09) should result in more restrictive requirements for individual specifications.

(2) TECHNICAL SPECIFICATION(S): 3.3.1, "Reactor Protection System Instrumentation," Table 3.3.1-1, ACTION 3 and ACTION 9.

PAGE NUMBER(S): 3/4 3-4

DESCRIPTION OF PROPOSED CHANGE(S): Remove footnote indicator "*" to ACTION 3 and ACTION 9 of Table 3.3.1-1 and delete the corresponding footnote as indicated on marked up Technical Specification page 3/4 3-4 contained in Attachment 2, page 4 of 18.

JUSTIFICATION FOR PROPOSED CHANGE(S): ACTION 3 and ACTION 9 to Reactor Protection System Instrumentation Table 3.3.1-1 require the suspension of all operations involving CORE ALTERATIONS. Footnote "*" explains that the replacement of local power range monitor (LPRM) strings need not be suspended if these actions are entered.

The subject note creates the erroneous inference that replacement of LPRM strings is a CORE ALTERATION, which creates unnecessary confusion. Replacement of incore detectors, including LPRMs, from under the reactor vessel is not a CORE ALTERATION. Deletion of this note will eliminate such confusion.

An examination of the history of the Core Alteration definition clearly shows that replacement of incore instruments from undervessel, including LPRMs, is not a Core Alteration for the Perry Nuclear Power Plant, and that this is documented on the PNPP docket. The current PNPP Technical Specification definition of CORE ALTERATION was developed during preparation of the BWR-6 Standard Technical Specifications (STS) and was modified slightly for the PNPP Technical Specifications. During the development of the STS, the sentence that identifies that "normal movement" of the SRM's, IRM's, TIP's or special movable detectors is not considered a CORE ALTERATION was added, based on the fact that these incore instruments have a negligible impact on core reactivity. The normal movement of SRM's, IRM's, and TIP's includes complete withdrawal from the core area to their storage locations in the undervessel area, where they have absolutely no impact on reactivity and where their replacement also has no impact on reactivity. Replacement of SRMs, IRMs and TIPs is therefore not considered a Core Alteration (due to the existence of the dry tubes, withdrawal from the core/replacement of incore instruments also does not affect vessel integrity or create the possibility of damaging fuel).

This concept that "normal movement" includes undervessel replacement for the SRM's, IRM's, etc. was directly expanded to the LPRM's on the PNPP docket by a change made to the Core Alteration definition between low power and full-power licensing (reference Technical Specification Change Request Letter PY-CEI/NRR-0496L dated July 18, 1986). The July 18, 1986 change request noted that although the LPRM's have no normal drive mechanism, they are also removed from the core for replacement, and therefore the exception in the definition should also apply to them.

The July 18, 1986 change request also pointed out that LPRM replacement had previously been excepted from the definition of Core Alteration in a Technical Specification 3.3.1 footnote (the very footnote "*" for which deletion is herein requested).

The concept that replacement of incore LPRM strings from undervessel is not a Core Alteration by definition was approved for PNPP by issuance of the requested change to the definition of CORE ALTERATION in the Technical Specifications in PNPP's Full Power Operating License.

This concept was documented even more clearly in subsequent NRC correspondence when other plants such as River Bend and Clinton were revising their Technical Specifications to clarify that LPRM replacement is not a Core Alteration (reference River Bend Technical Specification Change Request Letter RBG-28400, File No. G9.5, G9.42, dated August 5, 1988 and Clinton Technical Specification Change Request Letter U-601460, LS-87-001, dated June 30, 1989).

The River Bend correspondence specifically cited PNPP as the precedent, but went into greater detail as to why replacement of incore instruments is considered a normal movement, and more fully described the dry tube concept which was an improvement over earlier GE BWR's.

The River Bend correspondence also requested the elimination of an identical footnote "*" to ACTION 3 and ACTION 9 in their Table 3.3.1-1, because with LPRM replacement not considered a CORE ALTERATION by definition, the note exempting LPRM replacement as a CORE ALTERATION was no longer necessary. These changes to the River Bend Technical Specifications were approved by the NRC in October 1988.

When Clinton filed their change request to add LPRM's to the definition of Core Alteration, they chose to add the words "including undervessel replacement" to the sentence that discusses movement of incore instruments, to make it clear what had been implicit in the licensing correspondence on PNPP and River Bend. The Clinton Technical Specifications never contained the footnote "*" in their Table 3.3.1-1. This was approved by the NRC staff in February 1990.

With LPRM replacement not considered a CORE ALTERATION by definition, the footnote "*" to ACTION 3 and ACTION 9 of Table 3.3.1-1 exempting LPRM replacement as a CORE ALTERATION under certain conditions is no longer necessary. Furthermore, retention

of the subject footnote has resulted in unnecessary confusion as to the status of LPRM string replacement as a Core Alteration. As stated above, deletion of this note will eliminate such confusion.

(3) TECHNICAL SPECIFICATION(S): Specification 3.3.7.5, "Accident Monitoring Instrumentation," Table 3.3.7.5-1, Item 2, Reactor Vessel Water Level, and associated Surveillance Requirement 4.3.7.5, Table 4.3.7.5-1, Item 2, Reactor Vessel Water Level.

PAGE NUMBER(S): 3/4 3-78 and 3/4 3-80

DESCRIPTION OF PROPOSED CHANGE(S): Divide Table 3.3.7.5-1, Item 2, and Table 4.3.7.5-1, Item 2 (Reactor Vessel Water Level), into subitems (a) for Fuel Zone and (b) for Wide Range water level indicators as indicated on marked up Technical Specification pages 3/4 3-78 and 3/4 3-80 contained in Attachment 2, pages 5 and 6 of 18.

JUSTIFICATION FOR PROPOSED CHANGE(S): Regulatory Guide 1.97, Revision 2, contains a recommendation that the BWR Accident Monitoring reactor coolant level instrumentation have a range from the bottom of the core support plate to the centerline of the main steam line. To meet this guidance, PNPP proposed (in FSAR/USAR Table 7.1-4) a design with Fuel Zone instruments covering the range from -150" [150" below top of active fuel (TAF)] to 50" above TAF, and Wide Range instruments covering the range from 5" to 230" above TAF. This range of level instrumentation was acceptable to the NRC staff in their Safety Evaluation Report, Supplement 6, Section 7.5.2.2 (see also letter PY-CEI/NRR-1012L dated May 26, 1989). To meet the above commitment, and to satisfy the intent of Specification 3.3.7.5, both wide range and fuel zone instrumentation should be OPERABLE. In its current format, Specification 3.3.7.5 does not make the operability requirements for the individual Wide Range and Fuel Zone instrumentation clear. The purpose of the proposed change to Table 3.3.7.5-1, Item 2, is to provide the necessary clarification.

In order to meet the Reactor Vessel Level Accident Monitoring requirements, the required number of channels should be 2 Wide Range and 2 Fuel Zone instruments. Likewise, for the minimum channels Operable requirement, one Wide Range and one Fuel Zone instrument must be available.

Note that the above justification also applies to the proposed change to Technical Specification Table 4.3.7.5-1, Accident Monitoring Instrumentation Surveillance Requirements, Item 2, Reactor Vessel Water Level.

(4) TECHNICAL SPECIFICATION(S): 3.4.4, "Chemistry," ACTION c.

PAGE NUMBER(S): 3/4 4-13

DESCRIPTION OF PROPOSED CHANGE(S): Add OPERATIONAL CONDITION 2 to the last sentence of Specification 3.4.4, ACTION c, as indicated on marked up Technical Specification page 3/4 4-13 contained in Attachment 2, page 7 of 18.

JUSTIFICATION FOR PROPOSED CHANGE(S): The purpose of the proposed change is to provide clarification to Specification 3.4.4, ACTION c. The intent of the last sentence of ACTION c is to ensure that, in the event an engineering evaluation is relied on to justify continued plant startup from Operational Condition 4 or 5, with an out-of-limit conductivity, pH or chloride concentration, that the engineering evaluation is completed and the effects on the structural integrity of the reactor coolant system are determined acceptable for continued operation prior to entering a higher mode of operation.

Under the existing wording of ACTION c, a mode change to Operational Condition 3 is not allowed until the engineering evaluation is performed and an acceptable determination obtained. However, the ACTION statement fails to extend this requirement to mode changes to Operational Condition 2. It is typical for a BWR, during the performance of a plant startup, to move from Operational Condition 4 to Operational Condition 2, without entering Operational Condition 3 at any time. Therefore, the proposed change to ACTION c will make it clear that mode changes into either Operational Condition 2 or 3 are not allowed until it is determined that the structural integrity of the reactor coolant system remains acceptable for continued operation.

(5) TECHNICAL SPECIFICATION(S): 3.6.1.9 "Feedwater Leakage Control System."

PAGE NUMBER(S): 3/4 6-14

DESCRIPTION OF PROPOSED CHANGE(S): Revise the ACTION Statement for Specification 3.6.1.9 to read as follows: "With one FWLC system subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours." Refer to Attachment 2, page 8 of 18, for a copy of marked up Technical Specification page 3/4 6-14.

JUSTIFICATION FOR PROPOSED CHANGE(S): The proposed change to the existing wording of the ACTION Statement for Specification 3.6.1.9 corrects an existing typographical error which will in turn clarify the appropriate action to be taken under the Limiting Condition For Operation. The proposed change will make the ACTION statement consistent with its original intent and with that of other standard Technical Specification ACTION statements.

(6) TECHNICAL SPECIFICATION(S): 3.6.5.2, "Containment Humidity Control," Figure 3.6.5.2-1, "Containment Average Temperature vs. Relative Humidity."

PAGE NUMBER(S): 3/4 6-43

DESCRIPTION OF PROPOSED CHANGE(S): In Specification 3.6.5.2, Figure 3.6.5.2-1, the line which divides the regions of acceptable versus unacceptable operation is extrapolated to continue down in a linear manner such that it intersects the 0% relative humidity line at approximately 62°F. Refer to Attachment 2, page 9 of 18, for a marked up copy of the proposed change to Technical Specification page 3/4 6-43.

JUSTIFICATION FOR PROPOSED CHANGE(S): Figure 3.6.5.2-1, which shows the humidity levels which are considered acceptable for periods when containment integrity is required, is not clear on what humidity levels are acceptable for containment average air temperatures below 72°F. The proposed change will provide this clarification.

The intent of the subject figure is to show initial relative humidities at various containment temperatures which are acceptable in order to maintain peak vacuum inside containment $\leq$ 0.72 psi (design is $\leq$ 0.80 psi) following initiation of both containment spray loops. Based upon the results of an Engineering review of applicable calculations, it is conservative to assume that the boundary is a straight line extending to temperatures lower than that shown on the figure. Therefore, the line which divides the regions of acceptable versus unacceptable operation may be extrapolated to continue down in a linear manner such that it intersects the 0% relative humidity line at approximately 62°F. Operation above this line (down to the containment design temperature of 60°F) is considered acceptable.

(7) TECHNICAL SPECIFICATION(S): 3/4.6.6.1 "Secondary Containment Integrity," Surveillance Requirement 4.6.6.1.a.

PAGE NUMBER(S): 3/4 6-45

DESCRIPTION OF PROPOSED CHANGE(S): Revise Surveillance Requirement 4.6.6.1.a to require verifying that the vacuum within the secondary containment is greater than or equal to 0.66 inches of vacuum water gauge instead of 0.40 inches of vacuum water gauge as it currently reads. Refer to Attachment 2, page 10 of 18, for a copy of marked up Technical Specification page 3/4 6-45.

JUSTIFICATION FOR PROPOSED CHANGE(S): The change to Surveillance Requirement 4.6.6.1.a is proposed in response to NRC Information Notice No. 88-76, "Recent Discovery Of A Phenomenon Not Previously Considered In The Design Of Secondary Containment Pressure Control," dated September 19, 1988. The Information Notice described a situation where required secondary containment (annulus) differential pressure (delta-P) was not met due to temperature

differences between outside air and annulus air, and sensor locations being 170 feet below the top of secondary containment.

In response to the NRC's Information Notice, a review of applicable PNPP design criteria was performed. The results of the design review revealed that outside air temperature was not previously considered and that most sensors are located approximately 166 feet below the top of secondary containment. PNPP's design bases and safety analysis require the secondary containment (annulus area) to be maintained at a minimum negative pressure of 0.25 inches water gauge at all times. PNPP's existing Technical Specification 4.6.6.1.a setpoint of 0.40 inches vacuum water gauge was established to maintain this minimum negative pressure of 0.25 inches water gauge, even during post-LOCA conditions. Based on engineering calculations performed in response to the Information Notice, the new analytical setpoint required to maintain the minimum negative pressure of 0.25 inch vacuum water gauge was determined to be 0.654 inches water gauge post LOCA and 0.50 inches water gauge during normal minimum design environmental conditions. Calculations to establish new field setpoints for delta-P and for secondary containment air flow values based on the new analytical setpoint were subsequently completed and the new field setpoints were installed in the plant.

The design approach used to recalculate the setpoint for the delta-P instrumentation remained consistent with the original design basis and safety analysis and accounted for the phenomenon described in NRC Information Notice No. 88-76 with adjustments for specific conditions at the Perry Plant. Subsequently, PNPP's USAR sections 6.5.3.2.1.b, 6.5.3.2.2, 6.5.3.2.3 and 7.3.1.1.9.b were revised during the 1989 USAR update to meet the design basis for secondary containment negative pressure in reference to NRC Information Notice No. 88-76. The revised delta-P and airflow values permit Perry's Annulus Exhaust Gas Treatment (M15) system to operate and maintain secondary containment integrity as described in Perry's USAR.

This proposed change to PNPP Technical Specifications updates Surveillance Requirement 4.6.6.1.a to reflect the new analytical setpoint required to maintain secondary containment minimum negative pressure of 0.25 inches vacuum water gauge and to thereby meet PNPP's design basis for secondary containment negative pressure in reference to NRC Information Notice No. 88-76. The requested Technical Specification setpoint change remains consistent with PNPP's original USAR design bases and safety analysis. The change in the setpoint for negative pressure maintained in secondary containment is a change in the conservative direction and is administratively controlled to meet the intent of the Technical Specification. Furthermore, system function remains within the parameters originally specified and is not changed by this proposed change to Specification 4.6.6.1.a. Therefore, there is no change to the margin of safety as specified in the PNPP Technical Specifications as a consequence of the proposed change.

(8) TECHNICAL SPECIFICATION(S): Specification 3.8.1.1, "A.C. Sources-Operating," ACTION e.

PAGE NUMBER(S): 3/4 8-2

DESCRIPTION OF PROPOSED CHANGE(S): In Specification 3.8.1.1, Action e, replace the words "... in addition to ACTION b or c, as applicable, ..." with "... in addition to ACTION b, c, or g**, as applicable,..." and add the following footnote: "**When either the Div 1 or Div 2 diesel is restored to OPERABILITY." Refer to Attachment 2, page 11 of 18, for a marked up copy of Technical Specification page 3/4 8-2.

JUSTIFICATION FOR PROPOSED CHANGE(S): The proposed change is requested to clarify the correct ACTION(s) to be taken in the event that both the Division 1 and Division 2 diesel generators are declared inoperable requiring entry into ACTION g, followed by one of the inoperable diesel generators being restored to OPERABLE status while the other inoperable diesel generator remains inoperable.

Under the above scenario, ACTION g contains the requirements and time constraints for returning first one and then both of the inoperable Division 1 and 2 diesel generators to operability. Therefore, ACTION g is followed until both diesel generators are restored to operability.

In addition, although not specifically required by ACTION g, it is appropriate to perform ACTION e upon restoration of the first inoperable diesel generator to operability. ACTION e requires that within 2 hours a verification be made for all systems, subsystems, trains, components, and devices that depend on the restored diesel generator as a source of emergency power. Therefore, this Technical Specification change will make it clear that ACTION e should be entered upon restoration of the first of the two inoperable diesel generators.

In summary, once the first of two inoperable diesel generators is restored to OPERABLE status, the requirements of ACTION g should continue to be followed, and the requirements of ACTION e for the restored diesel generator should be implemented. The proposed change will clarify these requirements for the above scenario.

(9) TECHNICAL SPECIFICATION(S): Specification 3.9.12.d, "Inclined Fuel Transfer System" and associated Surveillance Requirement 4.9.12.2.a.

PAGE NUMBER(S): 3/4 9-18 and 3/4 9-19

DESCRIPTION OF PROPOSED CHANGE(S): Revise Limiting Condition For Operation (LCO) 3.9.12.d and Surveillance Requirement (SR) 4.9.12.2.a to read as follows: "At least one IFTS carriage position indicator is OPERABLE at each carriage position and at least one liquid level sensor is OPERABLE at each liquid level monitoring

position." Refer to Attachment 2, pages 12 and 13 of 18, for a copy of marked up Technical Specification pages 3/4 9-18 and 3/4 9-19.

JUSTIFICATION FOR PROPOSED CHANGE(S): The proposed changes are intended to provide clarification to LCO 3.9.12.d and SR 4.9.12.2.a. As discussed in the Bases for Specification 3.9.12, the purpose of the Inclined Fuel Transfer System (IFTS) specification is to control personnel access to those potentially high radiation areas immediately adjacent to the system and to assure safe operation of the system. Specification 3.9.12.d and its associated surveillance requirements are intended to verify Operability of the proximity and liquid level sensors associated with the Inclined Fuel Transfer System.

PNPP's Inclined Fuel Transfer System contains twelve (12) separate carriage positions with redundant (2) proximity sensors provided at each carriage position (total 24 proximity sensors with 2 sensors at each of the twelve carriage positions). The Technical Specification requirement in the "first clause" of Specification 3.9.12.d is to have at least 1 of the 2 redundant sensors for each of these twelve carriage positions be Operable. However, the present wording contained in the first clause of Specification 3.9.12.d fails to make this clear in that it requires at least one IFTS carriage position indicator be Operable "... at each of the twelve proximity sensors ..." The proposed change, which is consistent with the Technical Specifications of other BWR-6s with similar IFTS designs, will make this requirement clear, in that it would require at least one IFTS carriage position indicator be Operable "at each carriage position," i.e., at each of the twelve proximity sensors locations.

The proposed change also provides clarification to the second clause of LCO 3.9.12.d and SR 4.9.12.2.a. Perry's IFTS contains two (2) liquid (water) level monitoring locations or positions (the "Tube Full" and the "Tube Empty" positions) with redundant (2) liquid level sensors at each monitoring position (total 4 liquid level sensors). The second clause of Specification 3.9.12.d is intended to require Operability of at least one of the two redundant liquid level sensors at each monitoring location.

However, the present wording contained in the second clause of Specification 3.9.12.d fails to make this clear in that it merely requires "... at least one liquid level sensor ..." be operable. The proposed change will make this requirement clear in that it would require at least one liquid level sensor "at each liquid level monitoring position" be Operable.

(10) TECHNICAL SPECIFICATION(S):

TECHNICAL SPECIFICATION(S):	PAGE NUMBER(S):
4.7.4.e, Snubbers	3/4 7-10
6.7.1.c, Safety Limit Violation	6-15
6.9.1, Routine Reports	6-17
6.9.1.8, Monthly Operating Reports	6-21
6.9.1.9, Core Operating Limits Report	6-21

6.9.2, Special Reports	6-21a
6.9.3, Special Reports	6-21a
6.9.4, Special Reports	6-21a

DESCRIPTION OF PROPOSED CHANGE(S): The proposed changes revise the PNPP Technical Specifications by changing those Specifications involving written reports submitted to the NRC in order to be consistent with the written communication requirements of 10 CFR 50.4, "Written Communications." Refer to Attachment 2 pages 14, 15, 16, 17 and 18 of 18, for a marked up copy of the subject Technical Specification pages listed above.

JUSTIFICATION FOR PROPOSED CHANGE(S): The proposed changes would revise the Technical Specifications by changing those specifications involving written reports submitted to the NRC in order to be consistent with the written communication requirements of amended rule 10 CFR 50.4, effective January 5, 1987 (51 FR 27817, August 4, 1986). The proposed changes are purely administrative in that they remove administrative inconsistencies between PNPP Technical Specifications and 10 CFR 50.4, where in Part 50.4(f) the Commission has clearly stated that Section 50.4 takes precedent over existing Technical Specifications. As stated by the Commission in the Statements of Consideration accompanying the publication of final amended rule 10 CFR 50.4 (51 FR 40303), this rule supersedes all existing requirements and guidance with respect to the number of copies and mailing procedures for submitting correspondence, reports, applications or other written communications pertaining to the domestic licensing of utilization facilities. And, licensees whose technical specifications contain conflicting submittal directions are authorized by this rule to delete the conflicting directions by pen-and-ink changes to their Technical Specifications.

These proposed changes simply allow for administrative incorporation of the changes that were codified by the 10CFR50.4 rulemaking. Being editorial in nature, the proposed changes have no impact on plant equipment or methods of operation.

II. SIGNIFICANT HAZARDS CONSIDERATION

The standards used to arrive at a determination that a request for amendment involves no significant hazards considerations are included in the Commission's Regulations, 10 CFR 50.92, which state that the operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (2) create the possibility of a new or different kind of accident from any previously evaluated, or (3) involve a significant reduction in a margin of safety.

The proposed amendment has been reviewed with respect to these three factors and it has been determined that the proposed changes do not involve a significant hazard because:

(1) The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated because the proposed changes constitute either (1) purely administrative changes designed to achieve consistency throughout PNPP Unit 1 Technical Specifications, provide clarification, correct existing errors, or delete material no longer applicable to PNPP Unit 1 Technical Specifications, (2) an additional limitation, restriction or control not presently included in the PNPP Unit 1 Technical Specifications, or (3) changes to conform PNPP Unit 1 Technical Specifications to changes in NRC regulations, where the license changes result in very minor changes to facility operations clearly in keeping with the regulations. Each of the proposed changes have been reviewed and determined to result in no significant changes to plant systems. The proposed changes have no significant effect on accident conditions or assumptions. The proposed changes do not significantly affect possible initiating events for accidents previously evaluated, or any system functional requirements.

The proposed changes to Specification 3.3.1, Reactor Protection System Instrumentation, ACTION a, Specification 3.3.2, Isolation Actuation Instrumentation, ACTION b, and Specification 3.3.3, Emergency Core Cooling System Actuation Instrumentation, Table 3.3.3-1 ACTION 38, are administrative in nature and are being made to correct the Specifications to be consistent with the guidance in Generic Letter 87-09 as it related to section 3.0.4 of the Technical Specifications, which was modified by Amendment 30 to PNPP's Unit 1 Facility Operating License. As such, the proposed changes do not affect any accident previously evaluated.

The proposed changes to Specification 3.3.1, Reactor Protection System Instrumentation, Table 3.3.1-1, ACTION 3 and ACTION 9, to remove the note which excepts the replacement of local power range monitor (LPRM) strings, are purely administrative changes designed to achieve consistency between the PNPP Technical Specification definition of CORE ALTERATION and Specification 3.3.1. Based upon the current definition of CORE ALTERATION, which exempts the replacement of LPRM's, the subject footnote is no longer applicable and its removal will provide clarification and thereby eliminate unnecessary confusion. Consequently,

the proposed changes to Specification 3.3.1 ACTION 3 and ACTION 9 do not result in an increase in the probability or consequences of any accident previously evaluated.

The proposed changes to Specifications 3.3.7.5, Accident Monitoring Instrumentation, Table 3.3.7.5-1, Item 2, Reactor Vessel Water Level, and associated Surveillance Requirement 4.3.7.5-1, Table 4.3.7.5-1, Item 2, Reactor Vessel Water Level, are intended for clarification only. Subdividing the Reactor Vessel Water Level instrumentation into "Fuel Zone" and "Wide Range" will provide clarification as to the number of channels required, the minimum number of channels required to be operable and applicable instrument surveillance requirements. The clarification is requested due to PNPP's method of satisfying commitments to Regulatory Guide 1.97, Revision 2, which requires BWR Accident Monitoring reactor coolant level instrumentation to have a range from the bottom of the core support plate to the centerline of the main steam line. PNPP employs a design with Fuel Zone instruments covering the range from -150" [150" below top of active fuel (TAF)] to 50" above TAF, and Wide Range instruments covering the range from 5" to 230" above TAF (reference PNPP USAR Table 7.1-4 and SER, Supplement 6, Section 7.5.2.2). To meet the above monitoring requirement, and to satisfy the intent of Specification 3.3.7.5, both wide range and fuel zone instrumentation should be OPERABLE. However, in its current format, Specification 3.3.7.5 does not make the operability requirements for the individual Wide Range and Fuel Zone instrumentation clear. The proposed changes to Tables 3.3.7.5-1 and 4.3.7.5-1 will provide the necessary clarification. Since the proposed changes are provided for clarification only and do not change current Technical Specification Limiting Conditions for Operation or Surveillance Requirements, the changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change to Specification 3.4.4 to add "OPERATIONAL CONDITION 2" to the last sentence of ACTION c is also for clarification purposes. ACTION c provides the required Action to be taken with reactor coolant system conductivity, pH and Chloride concentration out-of-limit while in OPERATIONAL CONDITIONS 4 or 5. ACTION c requires either (1) that the out-of-limit condition be restored to within acceptable limits, or (2) an engineering evaluation be performed to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system. In addition, ACTION c, as currently worded, explicitly prohibits a mode change into OPERATIONAL CONDITION 3 until it is first determined that the structural integrity of the reactor coolant system remains acceptable for continued operation. The intent of this restriction is to ensure that the structural integrity of the reactor coolant system remains acceptable for continued operation prior to any plant startup from OPERATIONAL CONDITIONS 4 or 5. However, ACTION c does not explicitly require the "determination of acceptability" prior to a mode change into OPERATIONAL CONDITION 2 from OPERATIONAL CONDITIONS 4 or 5. Since it is typical for a BWR during the performance of a plant startup to move from OPERATIONAL CONDITION 4 directly into OPERATIONAL CONDITION 2, without entering OPERATIONAL CONDITION 3 at any time, the proposed change will make it clear that such a change is prohibited until after the required acceptability determination is completed. Since the proposed change to Specification 3.4.1, ACTION c, is for clarification

only, and does not otherwise change the Specification 3.4.4 Action requirements, the change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

The proposed change to the wording of the ACTION statement for Technical Specification 3.6.1.9, Feedwater Leakage Control System, is a purely administrative change designed to correct an existing typographical error and in turn provide clarification of the appropriate action to be taken under the subject Specification's Limiting Condition For Operation. The proposed change will make the ACTION statement consistent with its original intent and with that of other standard Technical Specification ACTION statements.

The purpose of the proposed change to Figure 3.6.5.2-1, Containment Average Temperature vs. Relative Humidity, which provides an extension to the dividing line between regions of acceptable versus unacceptable operation, is to provide clarification on what humidity levels are acceptable for containment average air temperature below 72°F. The current Figure fails to provide meaningful operational limits below 72°F and 8% relative humidity, the point at which the dividing line terminates. The intent of Specification 3.6.5.2 is to restrict operation to within specified temperature versus relative humidity limits to prevent excessive vacuum from being created inside containment following an inadvertent initiation of the containment spray system. By maintaining containment average temperatures and relative humidities within the acceptable operational limits specified in Figure 3.6.5.2-1, peak vacuum inside containment will be maintained $\leq$ 0.72 psi (design is $\leq$ 0.80 psi) following initiation of both containment spray loops. Based on the results of an Engineering review of applicable calculations, it is conservative to assume that the boundary between acceptable operation is a straight line extending to tempe tures below that shown on the subject figure. Therefore, extending the line which divides the regions of acceptable versus unacceptable operation down in a linear manner such that it intersects the 0% relative humidity line at approximately 62°F will provide clarification on acceptable versus unacceptable operational limits below 72°F. Maintaining temperature and relative humidity within the clarified limits for acceptable operation below 72°F will continue to ensure peak vacuum inside containment will be maintained $\leq$ 0.72 psi following initiation of both containment spray loops. The design methodology used to provide the clarification to Figure 3.6.5.2-1 remains consistent with the original design bases and safety analysis. Therefore, the proposed change to Figure 3.6.5.2-1 will not involve a significant increase in the probability or consequences of any accident previously evaluated.

The change to the limit for secondary containment (annulus) minimum negative pressure contained in Technical Specification Surveillance Requirement 4.6.6.1.a is proposed in response to NRC Information Notice (IN) 88-76, "Recent Discovery Of A Phenomenon Not Previously Considered In The Design Of Secondary Containment Pressure Control," dated September 19, 1988. The change replaces existing secondary containment minimum negative pressure verification requirement of 0.40 inches of vacuum water gauge with 0.66 inches of vacuum water gauge. As such the change constitutes a more stringent surveillance requirement than that

previously required. The change is intended to ensure that the secondary containment minimum negative pressure of 0.25 inches water gauge required by PNPP's original design bases and safety analysis is maintained at all times. The design methodology used to recalculate the setpoint for the differential pressure (delta-P) instrumentation remains consistent with the original design bases and safety analysis and accounts for the phenomenon described in NRC Information Notice 88-76 with adjustments for specific conditions at the Perry Plant. The revised delta-P and airflow values will permit the M15 system to operate and maintain secondary containment integrity as described in PNPP's USAR. Based on the fact that overall system function has not changed, the parameters upon which the PNPP USAR safety analysis (USAR Chapter 15.6.5.5.1.2.a) was based having not been affected. Consequently, the proposed change to Surveillance Requirement 4.6.6.1.a does not involve a significant increase in the possibility or consequences of an accident previously evaluated.

The proposed change to Specification 3.8.1.1, A.C. Sources-Operating, ACTION e, is a purely administrative change designed to provide clarification as to the appropriate actions to be taken in the event both the Division 1 and Division 2 diesel generators are declared inoperable, requiring entry into ACTION g, followed by one of the inoperable diesel generators being restored to OPERABLE status, while the other remains inoperable. Consequently, the proposed change to Specification 3.8.1.1, ACTION e, does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Likewise, the proposed changes to Specification 3.9.12.d, Inclined Fuel Transfer System and associated Surveillance Requirement 4.9.12.2.a are purely administrative changes designed to provide clarification of the Limiting Conditions For Operation and the Surveillance Requirements associated with the Inclined Fuel Transfer System proximity and liquid (water) level sensors. As such, the proposed changes to Specification 3.9.12.d and Surveillance Requirement 4.9.12.2.a do not involve a significant increase in the probability or consequences of an accident previously evaluated.

Finally, the proposed changes to Specifications 4.7.4.e, Snubbers, 6.7.1.c, Safety Limit Violations, 6.9.1, Routine Reports, 6.9.1.8, Monthly Operating Reports, 6.9.1.9, Core Operating Limits Report, 6.9.2, Special Reports, 6.9.3, Special Reports and 6.9.4, Special Reports, are changes designed to conform the reporting requirements of the subject Specifications to changes in NRC regulation 10 CFR 50.4, Written Communications (reference 51 FR 27817, August 4, 1986). The proposed changes are purely administrative in that they are designed to remove administrative inconsistencies between PNPP Unit 1 Technical Specifications and 10 CFR 50.4 where the Commission has clearly stated that Section 50.4 takes precedent over existing Technical Specifications. The proposed changes have no impact on plant equipment or methods of PNPP facility operations and are clearly in keeping with amended rule 10 CFR 50.4. Therefore, the proposed changes to the reporting requirements of the subject Specifications cannot increase the probability or consequences of an accident previously evaluated.

Based upon the above, the subject technical and administrative changes proposed herein do not increase the probability or consequences of any accident previously evaluated.

(2) The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

As stated above, the proposed changes are either administrative in nature which do not increase the possibility of any new or different kind of accident, or constitute more conservative limitations, restrictions or controls than that presently included in PNPP Technical Specifications. The proposed changes do not create the possibility of a new or different kind of accident since they do not affect the reactor coolant pressure boundary or other plant systems or structures in such a manner that could initiate any new or different kind of accident. In addition, the proposed changes do not adversely affect any system functional requirements nor plant maintenance or operability requirements in such a manner that could initiate any new or different kind of accident. Consequently, no new failure modes are introduced as a result of the proposed changes.

(3) The proposed changes do not result in a significant reduction in the margin of safety.

The changes do not involve a significant reduction in the margin of safety because they are administrative in nature, and do not affect any USAR design bases or accident assumptions, or they constitute more conservative limitations, restrictions or controls than that presently included in PNPP Technical Specifications. Therefore, the proposed changes do not reduce the margin of safety as defined in the basis for any Technical Specification.

Based upon the above considerations, it has been concluded that the proposed changes do not involve significant hazards considerations.

ENVIRONMENTAL CONSIDERATION

The proposed Technical Specification changes have been reviewed against the criteria of 10 CFR 51.22 for environmental considerations. As shown above, the proposed changes do not involve a significant hazards consideration, nor do they increase the types and amounts of effluents that may be released offsite, nor do they significantly increase individual or cumulative occupational radiation exposures. Based on the foregoing, it has been concluded that the proposed Technical Specification changes meet the criteria given in 10 CFR 51.22(c)(9) for a categorical exclusion from the requirement for an Environmental Impact Statement.