August 19, 2008

Vice President, Operations
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SUBJECT: VERMONT YANKEE NUCLEAR POWER STATION - REQUEST FOR ADDITIONAL INFORMATION REGARDING INSTRUMENTATION TECHNICAL SPECIFICATION CHANGES (TAC NO. MD8111)

Dear Sir or Madam:

By letter dated February 12, 2008 (Agencywide Documents and Management System (ADAMS) Accession No. ML080660458), Entergy Nuclear Operations, Inc. (the licensee) requested changes to Vermont Yankee Nuclear Power Station (VY) Technical Specifications (TSs).

The Nuclear Regulatory Commission staff has been reviewing TS Sections 1.0, 1/2.1, 3/4.1, 3.2.A, and 3.2.B relating to reactor protection system and has determined that additional information is needed to complete its review. The specific questions are found in the enclosed request for additional information (RAI). A response to this RAI is requested to be provided within 30 days.

Sincerely,

/RA/

James Kim, Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-271

Enclosure:
As stated

cc w/encl: See next page

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VERMONT YANKEE - REQUEST FOR ADDITIONAL INFORMATION REGARDING

INSTRUMENTATION TECHNICAL SPECIFICATION (TS) CHANGES

1. Explain how the proposed Section 1.0 definition for Reactor Protection System (RPS) Response Time ensures that the individual channel response times are less than or equal to the maximum values assumed in the accident analysis as required per 10 CFR 50.36(d)(3).

   Background: The proposed Section 1.0 definition for the RPS Response Time is defined as "the time from the opening of the sensor contact up to and including the opening of the scram solenoid relay." The proposed definition is derived from the current wording found in Limiting Condition for Operation (LCO) 3.1.A, and does not include response time testing of the sensor. The actual performance of the RPS Response Time is required per proposed Surveillance Requirement (SR) 4.1.A.3.

   For comparison purposes, NUREG-1433, "Standard Technical Specifications (STS), General Electric Plants, BWR/4," LCO 1.1, "Definitions," defines RPS Response Time as "that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids." The STS definition includes response time testing of the sensor. STS LCO 3.3.1.1, "RPS Instrumentation," contains SR 3.3.1.1.15, RPS Response Time. The Bases for STS SR 3.3.1.1.15 state the SR ensures that the individual channel response times are less than or equal to the maximum values assumed in the accident analysis. The Bases do state, however, that neutron detectors are excluded from RPS Response Time testing because the principles of detector operation virtually ensure an instantaneous response time. The Bases also state that the sensors for certain other functions are allowed to be excluded from specific RPS Response Time measurement if the conditions of Reference 12 are satisfied. Reference 12 is NEDO-32291-A, "System Analyses for the Elimination of Selected Response Time Testing Requirements," October 1995.

   10 CFR 50.36(d)(3) states TSs will include SRs which "are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met."

   It is unclear how the proposed Section 1.0 definition for RPS Response Time, which does not include response time testing of the sensor, ensures that the individual channel response times, as required per proposed SR 4.1.A.3, are less than or equal to the maximum values assumed in the accident analysis, thereby ensuring that facility operation will be within safety limits as required per 10 CFR 50.36(d)(3).

2. Explain how 10 CFR 50.36(d)(1)(ii)(A) is met when the proposed LCO 2.1.A.1.b does not require that the Intermediate Range Monitor (IRM) flux scram setting be set at less than or equal to 120/125 of full scale when the reactor mode switch is in the Refuel position with reactor coolant temperature at less than or equal to 212 ^{o}F.

Enclosure

Background: Proposed LCO 2.1.A.1.b, "Flux Scram Trip Setting (Refuel or Startup/Hot Standby Mode)," states "when the reactor mode switch is in the Refuel position (with reactor coolant temperature > 212 ^{o}F) or the Startup/Hot Standby position, average power range monitor (APRM) scram shall be set down to less than or equal to 15% of rated neutron flux. The IRM flux scram setting shall be set at less than or equal to 120/125 of full scale." As a result, the IRM flux scram setting would not be required to be set at less than or equal to 120/125 of full scale when the reactor mode switch is in the Refuel position with reactor coolant temperature at less than or equal to 212 ^{o}F. This is in contrast to current and proposed requirements.

The proposed Bases for LCO 3.1 state "in Refuel with reactor coolant temperature ≤ 212 ^{o}F, when a cell with fuel has its control rod withdrawn, the IRMs provide monitoring for and protection against unexpected reactivity excursions." In addition, proposed LCO 2.1.A.1.b does not reflect current requirements as stated in Note 1 of Table 3.1.1 or proposed Note (b) of Table 3.1.1 of LCO 3.1, "Reactor Protection System (RPS)." Note 1 states that "when the reactor is subcritical and the reactor water temperature is less than 212 ^{o}F, only the following trip functions need to be operable: c) high flux IRM or high flux SRM in coincidence." In the License Amendment Request (LAR), Note 1 is modified, in part, to proposed Note (b). Note (b) requires that the IRM High Flux (Function 3.a) be Operable in Refuel "with reactor coolant temperature ≤ 212 ^{o}F and any control rod withdrawn from a core cell containing one or more fuel assemblies."

10 CFR 50.36(d)(1)(ii)(A) states that "limiting safety system settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded."

It is unclear how 10 CFR 50.36(d)(1)(ii)(A) is met when proposed LCO 2.1.A.1.b does not require that the IRM flux scram setting be set at less than or equal to 120/125 of full scale when the reactor mode switch is in the Refuel position with reactor coolant temperature at less than or equal to 212 ^{o}F.

3. Explain how 10 CFR 50.36(d)(3) is met when proposed SR 4.1.A.1 explicitly requires performance of proposed SR 4.1.A.2 and proposed SR 4.1.A.3, but not proposed SR 4.1.A.4.

   Background: Proposed SR 4.1.A.1 states that "RPS testing shall also be performed as indicated in Surveillance Requirements 4.1.A.2 and 4.1.A.3." However, proposed SR 4.1.A.1 does not mention testing requirements associated with proposed SR 4.1.A.4.

   10 CFR 50.36(d)(3) states TSs will include SRs which "are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met."

   It is unclear how 10 CFR 50.36(d)(3) is met when proposed SR 4.1.A.1 explicitly requires performance of proposed SR 4.1.A.2 and proposed SR 4.1.A.3, but not proposed SR 4.1.A.4.

4. Discussion of Change L.3 does not provide information to support modifying actions for when the APRM High Flux (flow bias) is inoperable, as indicated in the LAR. Please provide the justification supporting the requested changes to the APRM High Flux (flow bias) actions.

   Background: The current requirements for an inoperable APRM High Flux (flow bias) are found in Note 2 of Table 3.1.1 of LCO 3.1.1, "Reactor Protection System." If those requirements are not met, the actions of Note 3 are carried out. Per Table 3.1.1, parts A or B of Note 3 are applicable to the APRM High Flux (flow bias). Action A of Note 3 is to "initiate insertion of operable rods and complete insertion of all operable rods within four hours," while alternate Action B is to "reduce power level to IRM range and place mode switch in the "Startup/Hot Standby" position within eight hours."

   The LAR proposes to delete Action A of Note 3 for the APRM High Flux (flow bias) while changing, in part, Action B of Note 3 into proposed Note 2.b. The reasoning for these changes is stated to be listed under Discussion of Change (DOC) A.14 and L.3. DOC L.3, however, does not contain any discussions pertinent to the APRM High Flux (flow bias).

   10 CFR 50.36(d)(2)(i) states "limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met."

   It is unclear how DOC L.3 provides information to support modifying actions for when the APRM High Flux (flow bias) is inoperable, as permitted by 10 CFR 50.36(d)(2)(i).

5. Discussion of Change LA.10 does not provide information to support relaxing the stated testing frequency of the logic test for the RPS instrumentation trip functions. Please provide the justification for the requested change in testing frequency.

   Background: Proposed SR 4.1.A.4 states to "perform a Logic System Functional Test of RPS instrumentation Trip Functions once every Operating Cycle." This is a relaxation of the current testing frequency requirement found in Note 7 of Table 4.1.1. Note 7 states "a functional test of the logic of each channel is performed as indicated" in Table 4.1.1. The Instrument Channels listed in Table 4.1.1 have various Minimum Frequencies in which the functional test of the logic is performed. Most Minimum Frequencies are listed as every 3 months. The reasoning for these changes is stated to be listed under Discussion of Change (DOC) LA.10. DOC LA.10, however, does not contain any discussions pertinent to the stated relaxation in the testing frequency of the logic test for the RPS instrumentation trip functions.

   10 CFR 50.36(d)(3) states TSs will include SRs which "are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met."

   It is unclear how DOC LA.10 provides information to support relaxing the stated testing frequency of the logic test for the RPS instrumentation trip functions while ensuring that 10 CFR 50.36(d)(3) is still met.

6. Explain how the lack of specific applicable modes for the Emergency Core Cooling System trip functions will still ensure that 10 CFR 50.36(d)(2)(i) is met. In addition, provide specifics on when proposed Note (b) of Table 3.2.1 is applicable.

   Background: Table 3.2.1 of Limiting Condition for Operation (LCO) 3.2.A, “Emergency Core Cooling System (ECCS),” has a proposed Note (b) that states ECCS trip functions are applicable “when the associated ECCS subsystem is required to be operable.” The proposal is ambiguous in that no specific LCO is referenced in proposed Note (b). As a result, it is unclear when proposed Note (b) specifically applies.

   For comparison purposes, current LCO 3.2.A provides more specific information regarding ECCS trip function applicability and states “when the system(s) it initiates or controls is required in accordance with Specification 3.5, the instrumentation which initiates the emergency core cooling system(s) shall be operable in accordance with Table 3.2.1.” LCO 3.5 is titled “Core and Containment Cooling Systems,” and contains ECCS trip function applicability modes. In addition, NUREG 1433, “Standard Technical Specifications, General Electric Plants, BWR/4,” contains a Note (a) in Table 3.3.5.1-1 of LCO 3.3.5.1, “ECCS Instrumentation,” that is similar in intent to proposed Note (b). Note (a) states that ECCS trip functions are applicable “when associated ECCS subsystem(s) are required to be Operable per LCO 3.5.2, ‘ECCS - Shutdown.”

   10 CFR 50.36(d)(2)(i) states “limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility.”

   It is unclear how the lack of specific applicable modes for the ECCS trip functions will still ensure that lowest functional capability or performance levels of equipment required for safe operation of the facility are met per 10 CFR 50.36(d)(2)(i).

7. Explain how the diesel generator automatic start requirement on a high drywell pressure or reactor vessel low-low water level signals is captured in the TSs as required per 10 CFR 50.36(d)(2)(i).

   Background: Section 6.3 of the Updated Final Safety Analysis Report (UFSAR) states, in part, that the “high drywell pressure or reactor vessel low-low water level signals will also start emergency diesel generators to restore auxiliary power.” This discussion is also captured in the proposed Bases for LCO 3.2.A, “Emergency Core Cooling System (ECCS),” and states “automatic initiation of the Diesel Generators occurs for conditions of Low - Low Reactor Vessel Water Level or High Drywell Pressure.” The proposed Bases also state, in part, that “upon receipt of a Loss-of-Coolant Accident (LOCA) initiation signal, each Diesel Generator is automatically started.” However, it is not clear if this requirement is captured in the TS.

   For comparison purposes, NUREG-1433, “Standard Technical Specifications, General Electric Plants, BWR/4,” contains a Note (b) in Table 3.3.5.1-1 of LCO 3.3.5.1, “ECCS Instrumentation.” Note (b) states, in part, that certain ECCS trip functions are “required to initiate the associated diesel generator (DG).”

   10 CFR 50.36(d)(2)(i) states “limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility.

It is unclear if the DG automatic start requirement on a high drywell pressure or reactor vessel low-low water level signals is captured in the TS in order to ensure that the lowest functional capability or performance levels of equipment required for safe operation of the facility is met per 10 CFR 50.36(d)(2)(i).

8. Explain how 10 CFR 50.36(d)(3) is met when proposed SR 4.2.B does not contain Response Time testing of Primary Containment Isolation Valves.

   Background: The UFSAR, Section 14.6.3, "Loss of Coolant Accident," states "containment isolation is postulated to occur simultaneously with the pipe break." UFSAR Table 5.2.2, "Primary Containment System Penetrations and Associated Containment Isolation Valves," lists minimum closing rates for various Primary Containment Isolation Valves (PCIVs). In addition, the proposed Bases for LCO 3.2.B, "Primary Containment Isolation," states "primary containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a DBA." However, proposed SR 4.2.B, "Primary Containment Isolation," does not have any Response Time testing associated with the PCIVs. For comparison purposes, NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4," contains SR 3.3.6.1.8, Response Time testing, in LCO 3.3.6.1, "Primary Containment Isolation Instrumentation."

   10 CFR 50.36(d)(3) states TSs will include SRs which "are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met."

   It is unclear how 10 CFR 50.36(d)(3) is met when proposed SR 4.2.B does not contain Response Time testing of PCIVs.

9. Explain why proposed Note 2.d provides acceptable actions to be taken for an inoperable High-Pressure Coolant Injection (HPCI) or Reactor Core Isolation Cooling (RCIC) System, due to an inoperable HPCI or RCIC System Isolation Trip Function, as permitted by 10 CFR 50.36(d)(2)(i).

   Background: Proposed Note 2.d in Table 3.2.2 of LCO 3.2.B, "Primary Containment Isolation," states in part to "isolate the affected penetration flow path within 1 hour." Proposed Note 2.d is applicable to the HPCI System Isolation Trip Function and the RCIC System Isolation Trip Function. No other actions are directed to be taken for an inoperable HPCI or RCIC System (due to the isolated penetration flow path). Proposed Note 2.d is vague in that other actions are expected for an inoperable HPCI or RCIC System, even if the actions of proposed Note 2.d for an inoperable HPCI or RCIC System Isolation Trip Function are carried out.

   For comparison purposes, current Note 3 in Table 3.2.2 of LCO 3.2.B, "Primary Containment Isolation," states "close isolation valves in system and comply with Specification 3.5." Note 3 is applicable to the HPCI and RCIC System Isolation Trip Function and LCO 3.5, "Core and Containment Cooling Systems," directs actions for an inoperable HPCI and RCIC System. In addition, NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4," contains LCO 3.3.6.1, "Primary

Containment Isolation Instrumentation." Condition F is applicable to the HPCI and RCIC System Isolation Trip Function and states to "isolate the affected penetration flow path" within "1 hour." However NUREG-1433 also contains LCO 3.0.3. LCO 3.0.3 states "when an LCO is not met and the associated Actions are not met, an associated Action is not provided, or if directed by the associated Actions, the unit shall be placed in a Mode or other specified condition in which the LCO is not applicable." Therefore LCO 3.0.3 directs actions be taken for an inoperable HPCI or RCIC System in LCOs associated with those systems (due to the isolated penetration flow path), even though the actions for an inoperable HPCI or RCIC System Isolation Trip Function are carried out in LCO 3.3.6.1. The Vermont Yankee TS do not contain a LCO 3.0.3 such as that found in NUREG-1433. As a result, there is no requirement for other actions to be taken for an inoperable HPCI or RCIC System when the actions of proposed Note 2.d for an inoperable HPCI or RCIC System Isolation Trip Function are carried out.

10 CFR 50.36(d)(2)(i) states "limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met."

It is unclear why proposed Note 2.d provides acceptable actions to be taken for an inoperable HPCI or RCIC System, due to an inoperable HPCI or RCIC System Isolation Trip Function, as permitted by 10 CFR 50.36(d)(2)(i).

10. Explain how proposed Note (c) in Table 3.2.2 ensures that the lowest functional capability or performance levels of equipment required for safe operation of the facility are met per 10 CFR 50.36(d)(2)(i).

    Background: Current LCO 3.2.B, "Primary Containment Isolation," states "when primary containment integrity is required, in accordance with Specification 3.7, the instrumentation that initiates primary containment isolation shall be operable in accordance with Table 3.2.2." LCO 3.7, "Station Containment Systems," states, in part, that "primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212^{o}F and fuel is in the reactor vessel." As a result, the HPCI System and the RCIC System Low Steam Supply Pressure Isolation Trip Functions are required to be operable when the reactor is critical or when the reactor water temperature is above 212 ^{o}F and fuel is in the reactor vessel.

    Proposed Trip Functions 3.c (HPCI Low Steam Supply Pressure) and 4.f (RCIC Low Steam Supply Pressure) of Table 3.2.2 of LCO 3.2.B state that the functions are applicable in Run, Startup / Hot Standby, Shutdown, and Refuel. There is a proposed Note (c) associated with Startup / Hot Standby, Shutdown, and Refuel that states "with reactor steam pressure > 150 psig." The relaxation associated with proposed Note (c) is not discussed in any of the Discussion of Change (DOC) tables.

    For comparison purposes, NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4," contains LCO 3.3.6.1, "Primary Containment Isolation Instrumentation." The HPCI Low Steam Supply Pressure (STS Function 3.b) and the RCIC Low Steam Supply Pressure (STS Function 4.b) are required to be operable in Modes 1 (Run), 2 (Refuel or Startup / Hot Standby), and 3 (Shutdown > 212 ^{o}F). There is no discussion of applicability with respect to reactor steam pressure > 150 psig.

10 CFR 50.36(d)(2)(i) states “limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility.”

It is unclear how proposed Note (c) in Table 3.2.2 ensures that the lowest functional capability or performance levels of equipment required for safe operation of the facility are met per 10 CFR 50.36(d)(2)(i).

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