Entergy

Entergy Nuclear Northeast
Entergy Nuclear Operations, Inc.
Indian Point Energy Center
295 Broadway, Suite 1
P.O. Box 249
Buchanan, NY 10511-0249

January 8, 2002

Re: Indian Point Unit No. 2
Docket No. 50-247
NL 02-001

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop O-P1-17
Washington, D.C. 20555-0001

SUBJECT: Indian Point Nuclear Generating Unit No. 2 License Amendment Request (LAR No. 02-001) - Deletion of Technical Specifications for Reactor Vessel Material Surveillance Program

Pursuant to 10CFR50.90, Entergy Nuclear Operations, Inc. (ENO) requests an amendment to the Indian Point Nuclear Generating Unit No. 2 (IP2) Technical Specifications (TS) to delete the requirements governing the reactor vessel material surveillance program in IP2 TS 3.1.B, "Heatup and Cooldown." Changes are also requested for TS Sections 4.2, “Inservice Inspection and Testing,” 5.2.C, “Design Features – Containment,” and 6.4, “Administrative Controls – Training,” to correct errors. In addition, changes are proposed for TS Sections 6.1, “Responsibility,” and 6.2, “Organization,” to reflect the organizational changes resulting from the license transfer to ENO.

Attachment 1 to this letter provides the description and evaluation of the proposed change. The revised TS pages are provided in Attachment 2 (strikeout/shadow format).

ENO requests that the proposed changes be approved by June 30, 2002 with an effective date within 60 days of approval.

The Station Nuclear Safety Committee (SNSC) and the Nuclear Facilities Safety Committee (NFSC) have reviewed the proposed changes. Both committees concur that the proposed changes do not represent a significant hazards consideration as defined by 10CFR50.92(c).

In accordance with 10CFR50.91, a copy of this submittal with its associated attachments is being submitted to the designated New York State official.

This submittal contains a new commitment that is provided in Attachment 3.

A001

Should you have any questions or require additional information, please contact Mr. John F. McCann, Manager Nuclear Safety and Licensing, at (914) 734-5074.

Sincerely,

Fred Dacimo
Vice President – Operations
Indian Point 2

Attachments

Cc: See page 3

cc:

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of Entergy Nuclear Operations, Inc. (Indian Point Station, Unit No. 2)	)))	Docket No. 50-247

APPLICATION FOR AMENDMENT TO
OPERATING LICENSE

Pursuant to Section 50.90 of the Regulations of the Nuclear Regulatory Commission, Entergy Nuclear Operations, as holder of Facility Operating License No. DPR-26, hereby applies for amendment of the Indian Point Nuclear Generating Unit No. 2 Technical Specifications contained in Appendix A of the license.

The specific proposed Technical Specification revisions are set forth in Attachment 2. The associated assessments demonstrate that the proposed changes do not involve a significant hazards consideration as defined in 10CFR50.92(c).

As required by 10CFR50.91(b)(1), a copy of this Application and an analysis concluding that the proposed changes do not involve a significant hazards consideration have been provided to the appropriate New York State official designated to receive such amendments.

BY:

Fred Dacimo
Vice President – Operations
Indian Point 2

Subscribed and sworn to
before me this 8th day
January, 2002.

Karen L. Lancaster
Notary Public

KAREN L. LANCASTER
Notary Public, State of New York
No. [illegible]
Qualified In Westchester County
Term Expires 9/30/05

ATTACHMENT 1 TO NL 02-001

LICENSE AMENDMENT REQUEST

ENTERGY NUCLEAR OPERATIONS, INC
INDIAN POINT UNIT NO. 2
DOCKET NO. 50-247

LICENSE AMENDMENT REQUEST

DESCRIPTION OF THE PROPOSED CHANGE

Entergy Nuclear Operations, Inc. (ENO) is requesting a change to the Indian Point Nuclear Generating Unit No. 2 (IP2) Technical Specifications (TS) to delete the requirements for the reactor vessel material specimen withdrawal schedule. The TS that is affected by the proposed change is Section 3.1.B, "Heatup and Cooldown."

In accordance with the guidance of Generic Letter (GL) 91-01, "Removal of the Schedule for the Withdrawal of Reactor Vessel Material Specimens from Technical Specifications," ENO commits to maintaining the NRC-approved version of the reactor vessel material surveillance specimen withdrawal schedule in the UFSAR.

ENO also requests other changes to the following TS Sections:

- 4.2, "Inservice Inspection and Testing," to change a regulation reference.
- 5.2.C, "Design Features" – "Containment Systems," since sodium hydroxide is no longer used.
- 6.1, "Responsibility," and 6.2, "Organization," to reflect the organizational changes resulting from the license transfer to ENO.
- 6.4, "Administrative Controls" – "Training" to remove the reference to Appendix A to 10CFR Part 55.

REASON FOR THE PROPOSED CHANGE

The provisions of TS 3.1.B.2 and 3.1.B.3 for the periodic development of heatup, cooldown, and RCS integrity testing limits using the results of the reactor vessel material surveillance program duplicate requirements of 10CFR50 Appendix G, "Fracture Toughness Requirements," and 10CFR50 Appendix H, "Reactor Vessel Material Surveillance Program Requirements." This License Amendment Request was prepared using the guidelines of GL 91-01. GL 91-01 is directly applicable to the deletion of the reactor vessel material surveillance specimen removal schedule.

The following changes correct errors that exist in the TS:

- TS 4.2.1 incorrectly states that the inservice testing specification is required by 10CFR50.55a(g). The correct regulation is 10CFR50.55a(f).
- IP2 no longer uses sodium hydroxide to meet the requirements of TS 3.3.B, "Containment Cooling and Iodine Removal Systems."
- TS 6.4 requires a retraining and replacement training program that meets or exceeds the requirements of Appendix A to 10CFR Part 55. Appendix A to 10CFR Part 55 has been deleted from 10CFR.

Changes to TS Sections 6.1, "Responsibility," and 6.2, "Organization," reflect the organizational change resulting from the license transfer to ENO.

EVALUATION OF THE PROPOSED CHANGE

The Pressure/Temperature (P/T) limits of TS sections 3.1.B.1 and 4.3.c are derived from the analyses and evaluations included in the safety analysis report. They are limiting conditions of operation that satisfy criterion 2 of 10CFR50.36 since they preclude non-ductile failure of the RCS, an unanalyzed condition. The limits provide an acceptable range of operating temperatures and pressures for heatup, cooldown, and inservice leak and hydrostatic testing.

10CFR50.60, "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation," requires ENO to meet the fracture toughness and material surveillance program requirements for the IP2 RCS pressure boundary that are set forth in 10CFR50 Appendix G, "Fracture Toughness Requirements," and Appendix H, "Reactor Vessel Material Surveillance Program Requirements." Compliance with 10CFR50.60 is a condition of the IP2 Facility Operating License.

10CFR50 Appendix G specifies the analysis method that must be used to determine reactor vessel P/T limits. The analysis must account for the effects of neutron radiation including the results of the surveillance program of 10CFR50 Appendix H. 10CFR50 Appendix H requires prior NRC approval of changes to the reactor vessel surveillance specimen withdrawal schedule.

The reactor vessel materials surveillance program is described in detail in UFSAR section 4.5.2.

These proposed changes are consistent with NUREG-1431, "Standard Technical Specifications Westinghouse Plants," Section 3.4.3, "RCS Pressure and Temperature (P/T) Limits," in that the Standard TS (STS) require compliance with P/T limits but do not specify the methods by which the limits are developed. The requirements for the methods are described in the STS Bases. Approval of the proposed change will facilitate the transition to STS at IP2.

By eliminating duplication, the proposed TS simplifies administrative processes both for ENO and the NRC. In addition, ENO has concluded that the proposed TS ensures a level of regulatory control that is equivalent to the current TS since compliance with 10CFR50.60 is assured. This ENO conclusion is consistent with the conclusion reached by the NRC in GL 91-01 that there would be no loss of regulatory control for the deletion of a TS when the requirements of the deleted TS duplicate the requirements of a regulation.

Evaluation of Other Changes

10CFR50.55a(g) is titled "Inservice Inspection Requirements." IP2 TS 4.2.2, "Inservice Inspection," requires implementation of 10CFR50.55a(g). 10CFR50.55a(f) is titled "Inservice Testing Requirements" which is the subject of TS 4.2.1, "Inservice Testing." This proposed change aligns the TS with the proper CFR paragraph.

In License Amendment 191 (Ref. 1), the NRC revised TS Sections 3.3 and 4.5 to allow the deletion of the requirement to use sodium hydroxide as an additive in the post-accident containment spray system. Therefore, the NRC has previously evaluated the requested change to TS Section 5.2.C. The pH control of containment spray to enhance iodine removal from the containment atmosphere is adequately described in the TS Bases for Section 3.3.

Since 10CFR55 no longer has an Appendix A, deletion of the reference to 10CFR55 Appendix A is appropriate. Operator retraining programs are now required by 10CFR55.59 and replacement operator training programs are explicitly subject to approval by the NRC in accordance with 10CFR55.31(a)(4). Since ENO is required by Facility Operating License DPR-26 to comply with all applicable regulations, specifically requiring compliance with 10CFR55 (or portions thereof) in the TS is duplicative. ENO has concluded that there will be no loss of regulatory control by not replacing the deleted regulation with the current regulations.

The proposed organizational changes reflect the current organization of ENO. The proposed wording is identical to that used in the comparable section in the Indian Point Nuclear Generating Unit No. 3 TS.

Conclusion

In conclusion, based on the considerations above, (1) there is a reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

NO SIGNIFICANT HAZARDS CONSIDERATION

The proposed changes described above do not involve a significant hazards consideration. This conclusion is based on the evaluation, in accordance with 10CFR50.91(a)(1), of the three standards set forth in 10CFR50.92(c).

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

   The proposed change to TS Section 3.1.B involves deleting specific TS requirements that duplicate the requirements of 10CFR50.60, 10CR50 Appendix G, and 10CFR50 Appendix H. The proposed change does not result in a change to the design or operation of any plant structure, system or component. Therefore any assumptions of the operability or performance of any structure, system or component in accident evaluations are unchanged.

   The proposed change to TS 4.2.1 simply corrects an improper reference to the CFR. There are no physical changes to IP2 or to the operation of any system, structure, or component.

The proposed change to TS 5.2.C makes the design feature description consistent with TS Limiting Condition for Operation 3.3.B wherein the requirements for the method of post-accident iodine removal are specified. Making the Design Feature consistent with the appropriate LCO has no effect on the assumptions and the results of the accident analyses.

TS sections 6.1, 6.2, and 6.4 are administrative controls. Changing an administrative control has no affect on accident analyses.

Therefore, there will be no increase in the probability or in the consequences of an accident previously evaluated.

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

The proposed change to TS Section 3.1.B does not affect the effectiveness of ENO's implementation of the requirements of 10CFR50.60 that ensure the reactor vessel continues to be protected against non–ductile failure.

There is no change to any system, structure, or component as a result of any of the proposed changes.

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

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

The proposed TS changes simplify the methods of controlling the schedule for the reactor vessel surveillance specimen withdrawal schedule in that a duplicative control is removed. The effectiveness of ENO compliance with 10CFR50.60 and 10CFR50 Appendices G and Appendix H is not adversely affected by this change. The level of regulatory control for the reactor vessel pressure/temperature limits is not changed.

The effectiveness of IP2's inservice testing program is not affected by the correction of the improper CFR reference in TS 4.2.1.

ENO is required to comply with 10CFR55 at IP2. The effectiveness of ENO's compliance with 10CFR55 is not affected by deleting the improper CFR citation from TS 6.4. Similarly, ENO's compliance with the IP2 license and the all applicable laws and regulations is not affected by the proposed changes to the TS sections for responsibility and organization.

The change to the Design Features to properly identify the method specified in TS 5.2.B for post-accident iodine removal does not affect the margin of safety.

This change does not affect any design function for or the operation of any plant structure, system, or component.

Therefore, the change does not affect does not result in a change to any of the safety analyses or any margin of safety.

CONCLUSION

The proposed changes do not involve a significant increase in the probability or in the consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a significant reduction in a margin of safety. Accordingly, these proposed changes do not involve a significant hazards consideration. The Station Nuclear Safety Committee (SNSC) and the Nuclear Facilities Safety Committee (NFSC) have reviewed the proposed changes. Both committees concur that the proposed changes do not represent a significant hazards consideration as defined by 10CFR50.92(c).

ENVIRONMENTAL ASSESSMENT

An environmental assessment is not required for the above proposed changes because the requested changes to the Indian Point Generating Station Unit 2 Technical Specifications conform to the criteria for "actions eligible for categorical exclusion," as specified in 10CFR51.22(c)(9). The requested changes will have no impact on the environment. The proposed changes do not involve a significant hazards consideration as discussed in the preceding section. The proposed changes do not involve a significant change in the types or significant increase in the amounts of any effluents that may be released offsite. In addition, the proposed changes do not involve a significant increase in individual or cumulative occupational radiation exposure.

REFERENCES

1. NRC letter to Con Edison, titled "Issuance of Amendment for Indian Point Nuclear Generating Unit No. 2 (TAC No. M96548)," dated April 23, 1997

ATTACHMENT 2 TO NL 02-001

TECHNICAL SPECIFICATION PAGES IN

STRIKEOUT/SHADOW FORMAT

Deleted text is shown as ~~strikeout~~.

Added text is shown as shaded.

ENTERGY NUCLEAR OPERATIONS, INC
INDIAN POINT UNIT NO. 2
DOCKET NO. 50-247

B. HEATUP AND COOLDOWN

Specifications

1. The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) averaged over one hour shall be limited in accordance with Figure 3.1.B-1 and Figure 3.1.B-2 for the service period up to 21.63 effective full-power years. The heatup or cooldown rate shall not exceed 100°F/hr.

   a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those present may be obtained by interpolation.

   b. Figure 3.1.B-1 and Figure 3.1.B-2 define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2. The limit lines shown in Figure 3.1.B-1 and Figure 3.1.B-2 shall be recalculated periodically using methods discussed in WCAP-7924A and WCAP-12796. ~~and results of surveillance specimen testing as covered in WCAP-7323[7] and as specified in Specification 3.1.B.3 below. The order of specimen removal may be modified based on the results of testing of previously removed specimens. The NRC will be notified in writing as to any deviations from the recommended removal schedule no later than six months prior to scheduled specimen removal.~~

3. ~~The reactor vessel surveillance program* includes six specimen capsules to evaluate radiation damage based on pre-irradiation and post-irradiation tensile and Charpy V notch (wedge open loading) testing of specimens.~~ DELETED

~~* Refer to UFSAR Section 4.5, WCAP-7323, and Indian Point Unit No. 2, "Application for Amendment to Operating License," sworn to on February 3, 1981.~~

~~The specimens will be removed and examined at the following intervals:~~

~~Capsule 1~~	~~End of Cycle 1 operation~~
~~Capsule 2~~	~~End of Cycle 2 operation~~
~~Capsule 3~~	~~End of Cycle 5 operation~~
~~Capsule 4~~	~~End of Cycle 8 operation~~
~~Capsule 5~~	~~End of Cycle 16 operation~~
~~Capsule 6~~	~~Spare~~

4. The secondary side of the steam generator shall not be pressurized above 200 psig if the temperature of the steam generator is below 70°F.

5. The pressurizer heatup and cooldown rates averaged over one hour shall not exceed 100°F/hr and 200°F/hr, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.

6. Reactor Coolant System integrity tests shall be performed in accordance with Section 4.3 of the Technical Specifications.

Basis

Fracture Toughness Properties

All components in the Reactor Coolant System are designed to withstand the effects of the cyclic loads due to reactor system temperature and pressure changes[(1)]. These cyclic loads are introduced by normal unit load transients, reactor trips, and startup and shutdown operation. The number of thermal and loading cycles used for design purposes are shown in Table 4.1-8 of the UFSAR. During unit startup and shutdown, the rates of temperature and pressure changes are limited. The maximum plant heatup and cooldown rate of 100°F per hour is consistent with the design number of cycles and satisfies stress limits for cyclic operation[(2)].

The reactor vessel plate opposite the core has been purchased to a specified Charpy V-notch test result of 30 ft-lb or greater at a Nil-Ductility Transition Temperature (NDTT) of 40°F or less. The material has been tested to verify conformity to specified requirements and a NDTT value of 20°F has been determined. In addition, this plate has been 100 percent volumetrically

The use of the composite curve in the cooldown analysis is necessary because system control is based on a measurement of reactor coolant temperature, whereas the limiting pressure is calculated using the material temperature at the tip of the assumed reference flaw. During cooldown, the 1/4 T vessel location is at a higher temperature than the fluid adjacent to the vessel I.D. This condition is, of course, not true for the steady-state situation. It follows that the ΔT induced during cooldown results in a calculated higher allowable K_{IR} for finite cooldown rates than for steady state under certain conditions.

Because operation control is on coolant temperature, and cooldown rate may vary during the cooldown transient, the limit curves shown in Figure 3.1.B-2 represent a composite curve consisting of the more conservative values calculated for steady state and the specific cooling rate shown.

Pressurizer Limits

Although the pressurizer operates at temperature ranges above those for which there is reason for concern about brittle fracture, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, 1965 Edition and associated Code Addenda through the Summer 1966 Addendum.

References

(1) Indian Point Unit No. 2 UFSAR, Section 4.1.5.

(2) ASME Boiler & Pressure Vessel Code, Section III, Summer 1965, N-415.

(3) Indian Point Unit No. 2 UFSAR, Section 4.2.5.

(4) WCAP-7924A, "Basis for Heatup and Cooldown Limit Curves," W. S. Hazelton, S.L. Anderson, S.E. Yanichko, April 1975.

(5) ASME Boiler and Pressure Vessel Code, Section III, 1974 Edition, Appendix G.

(6) ASTM E185-79, Surveillance Tests on Structural Materials in Nuclear Reactors.

(7) ~~WCAP-7323, "Consolidated Edison Company, Indian Point Unit No. 2 Reactor Vessel Radiation Surveillance Program," S.E. Yanichko, May 1969.~~ DELETED

(8) Final Report - SWRI Project No. 02-4531 - "Reactor Vessel Material Surveillance Program for Indian Point Unit No. 2 Analysis of Capsule T," E.B. Norris, June 30, 1977.

(9) Supplement to Final Report - SWRI Project No. 02-4531 - "Reactor Vessel Material Surveillance Program for Indian Point Unit No. 2 Analysis of Capsule T," E.B. Norris, December 1980.

4.2 INSERVICE INSPECTION AND TESTING

Applicability

Applies to the inservice inspection of Quality Group* A, B, and C components and the inservice testing of pumps and valves whose function is required for safety.

Objective

To provide assurance of the continued integrity and/or operability of those structures, systems, and components to which this specification is applicable.

Specifications

4.2.1 Inservice Testing

Inservice testing of pumps and valves whose function is required for safety shall be performed in accordance with the applicable edition and addenda of Section XI of the ASME Boiler and Pressure Vessel Code as required by 10 CFR 50, Section ~~50.55a(g)~~ 50.55a(f), except where specific written relief pursuant to 10 CFR 50, Section 50.55a has been granted.

4.2.2 Inservice Inspection

Inservice inspection of Quality Group* (* Quality Group classification is in accordance with Revision 3 of Regulatory Guide 1.26.) A, B, and C components shall be performed in accordance with the applicable edition and addenda of Section XI of the ASME Boiler and Pressure Vessel Code as Required by 10 CFR 50, Section 50.55a(g), except where specific written relief pursuant to 10 CFR 50, Section 50.55a has been granted.

4.2.3 Primary Pump Flywheels

The flywheels shall be visually examined at the first refueling. At each subsequent refueling, one different flywheel shall be examined by ultrasonic methods. The examinations schedules are shown in Table 4.2-1.

C. CONTAINMENT SYSTEMS

1. The containment vessel has an internal spray system which is capable of providing a distributed borated water spray of at least 2200 gpm. ~~During the initial period of spray operation, sodium hydroxide would be added to the spray water to increase the removal of iodine from the containment atmosphere~~[3].

2. The containment vessel has an internal air recirculation system which includes five fan-cooler units (centrifugal fans and water cooled heat exchangers), with a total heat removal capability of at least 308.5 MBtu/hr under conditions following a loss-of-coolant accident and at service water temperature of 95°F.[4]

References

(1) UFSAR Section 5.1.2.2
(2) UFSAR Section 5.1.4
(3) UFSAR Section 6.3
(4) UFSAR Section 6.4

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The ~~Vice President-Nuclear Power~~ corporate officer with direct responsibility for the plant shall be responsible for overall facility activities and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Plant Manager shall be responsible for facility operations and shall delegate in writing the succession to this responsibility during his absence.

6.2 ORGANIZATION

6.2.1 Facility Management and Technical Support

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the Quality Assurance Program Description (QAPD).

b. The Plant Manager shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

c. The ~~Vice President-Nuclear Power~~ corporate officer with direct responsibility for the plant shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

d. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for (1) the Operation Manager's and the Assistant Operation Manager's SRO license requirement which shall be in accordance with Technical Specification 6.2.2.h, and, (2) the Radiation Protection Manager who shall meet or exceed the minimum qualifications of Regulatory Guide 1.8, September 1975.

6.3.2 The Plant Manager shall meet or exceed the minimum qualifications specified for Plant Manager in ANSI N18.1-1971.

6.3.3 The Watch Engineer shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Nuclear Training Manager and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 ~~and Appendix A of 10 CFR Part 55~~.

6.4.2 DELETED

6.5 REVIEW AND AUDIT

6.5.1 The review and audit functions of the Station Nuclear Safety Committee (SNSC) and the Nuclear Facilities Safety Committee (NFSC) are described in the Quality Assurance Program Description (QAPD).

6.6 REPORTABLE EVENT ACTION

6.6.0 A Reportable Event is defined as any of the conditions specified in 10 CFR 50.73a(2).

6.6.1 The following actions shall be taken in the event of a Reportable Event

ATTACHMENT 3 TO NL 02-001

Commitments

ENTERGY NUCLEAR OPERATIONS, INC
INDIAN POINT UNIT NO. 2
DOCKET NO. 50-247

Commitments

No.	Commitment Description	Implementation Schedule
1.	The NRC-approved version of the reactor vessel material specimen removal schedule will be maintained in the UFSAR 4.5.2.	Upon implementation of the License Amendment