Entergy

Entergy Nuclear Northeast
Entergy Nuclear Operations, Inc.
Entergy Nuclear Indian Point 2, LLC
P. O. Box 249
Buchanan, NY 10511

October 8, 2003

Re: Indian Point Unit No. 2
Docket No. 50-247
NL-03-154

Document Control Desk
U.S. Nuclear Regulatory Commission
Mail Station O-P1-17
Washington, DC 20555-0001

Subject: Supplement 8 to the Indian Point 2 License Amendment Request for Conversion to Improved Standard Technical Specifications

Reference:
1) Entergy letter (NL-02-016) to NRC, "License Amendment Request (LAR 02-005) Conversion to Improved Standard Technical Specifications," dated March 27, 2002
2) Entergy letter (NL-02-092) to NRC, "Supplement 1 to the Indian Point 2 License Amendment Request for Conversion to Improved Standard Technical Specifications," dated July 10, 2002
3) Entergy letter (NL-03-035) to NRC, "Supplement 2 to the Indian Point 2 License Amendment Request for Conversion to Improved Standard Technical Specifications," dated February 26, 2003
4) Entergy letter (NL-03-081) to NRC, "Supplement 3 to the Indian Point 2 License Amendment Request for Conversion to Improved Standard Technical Specifications," dated May 19, 2003
5) Entergy letter (NL-03-107) to NRC, "Supplement 4 to the Indian Point 2 License Amendment Request for Conversion to Improved Standard Technical Specifications," dated June 26, 2003
6) Entergy letter (NL-03-116) to NRC, "Supplement 5 to the Indian Point 2 License Amendment Request for Conversion to Improved Standard Technical Specifications," dated July 15, 2003
7) Entergy letter (NL-03-127) to NRC, "Supplement 6 to the Indian Point 2 License Amendment Request for Conversion to Improved Standard Technical Specifications," dated August 6, 2003
8) Entergy letter (NL-03-137) to NRC, "Supplement 7 to the Indian Point 2 License Amendment Request for Conversion to Improved Standard Technical Specifications," dated September 11, 2003

A001

Dear Sir:

By letter dated March 27, 2002 (Reference 1), as supplemented by letters dated July 10, 2002, February 26, 2003, May 19, 2003, June 26, 2003, July 15, 2003, August 6, 2003, and September 11, 2003 (References 2, 3, 4, 5, 6, 7, and 8 respectively), Entergy Nuclear Operations, Inc. (Entergy) requested to amend the Indian Point 2 (IP2) Plant Operating License, Appendices A and B, "Technical Specifications." The proposed amendment converts the IP2 Current Technical Specifications (CTS) to Improved Technical Specifications (ITS) in accordance with NUREG 1431, "Standard Technical Specifications—Westinghouse Plants," Rev. 2.

This letter supplements References 1 through 8 for six pages of the submittal. The supplement incorporates final approved results of calculations associated with Reactor Protection System Interlocks and the 480 Volt bus Degraded Voltage Function Time Delays. The Bases were updated in Sections 3.7.1 (Main Steam Safety Valves) and 3.7.6 (Condensate Storage Tank) to reflect Amendment 237 (1.4 Percent Power Uprate). In addition, editorial changes were made to the Bases of ITS 3.3.1 and 3.3.2 to improve clarity. These six pages are included in Attachment 1, "Marked–up Pages Affected by Supplement 8," as hand written mark-ups to clearly reflect each change and in Attachment 2, "Clean Typed Pages Affected by Supplement 8," as final, typed pages with the mark-ups incorporated. All of the changes have been previously discussed with your staff.

The no significant hazards determination associated with References 1 through 8 required no revision as a result of the changes being incorporated by Supplement 8. Therefore, it remains Entergy's conclusion that the conversion of the Indian Point 2 Technical Specifications to Improved Technical Specifications involves no significant hazards consideration as defined by 10 CFR 50.92.

As previously discussed in Supplements 6 and 7 (References 7 and 8), there are currently no outstanding CTS license amendments requests (LARs) for IP2 and Entergy does not anticipate the submittal of any LARs that would require review and approval prior to IP2 ITS implementation.

As was done with References 1, 2, 3, 4, 5, 6, 7, and 8, the material constituting Supplement 8 to the ITS submittal is enclosed herewith on CD-ROM. The Supplement 8 CD-ROM includes the following:

- Information originally transmitted by References 1, 2, 3, 4, 5, 6, 7, and 8;
- A copy of this cover letter and attachments; and
- The IP2 CTS through Amendment 237.

This supplement and the enclosed CD-ROM were prepared in accordance with the guidance provided in NRC Regulatory Issue Summary 2001–05, "Guidance on Submitting Documents to the NRC by Electronic Information Exchange or on CD-ROM." In accordance with 10 CFR 50.91, a copy of this submittal and the associated attachments are being submitted to the designated New York State official.

There are no commitments contained in this letter.

Should you or your staff have any questions regarding this matter, please contact me at (914) 734-5336.

Sincerely,

William S. Blair
IP2 ITS Project Manager

Attachments
Enclosure

cc: see page 4

cc: (w/o attachments and enclosure)

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ATTACHMENT 1 TO NL-03-154

Marked–up Pages Affected by Supplement 8

Entergy Nuclear Operations, Inc.
Indian Point Unit No. 2
Docket No. 50-247

Table 3.3.1-1 (page 4 of 6)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
17. Reactor Protection System Interlocks					2.5 E-11
a. Intermediate Range Neutron Flux, P-6	2$^{(d)}$	2 trains	Q	SR 3.3.1.11 SR 3.3.1.13	≥ 2.65 E-10 amp
b. Low Power Reactor Trips Block, P-7	1	2 trains	R	SR 3.3.1.11 SR 3.3.1.13	NA
c. Power Range Neutron Flux, P-8	1	4	R	SR 3.3.1.11 SR 3.3.1.13	≤ 26.4% RTP
d. Power Range Neutron Flux, P-10	1,2	4	Q	SR 3.3.1.11 SR 3.3.1.13	≤ 10% RTP (set) ≥ 3.6% RTP (reset)
e. Turbine First Stage Pressure, P-7 input	1	2	R	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≤ 9.25% turbine power
18. Reactor Trip Breakers (RTBs)$^{(h)}$	1,2	2 trains	P	SR 3.3.1.4	NA
	3$^{(a)}$, 4$^{(a)}$, 5$^{(a)}$	2 trains	C	SR 3.3.1.4	NA
19. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms	1,2	1 each per RTB	S	SR 3.3.1.4	NA
	3$^{(a)}$, 4$^{(a)}$, 5$^{(a)}$	1 each per RTB	C	SR 3.3.1.4	NA
20. Automatic Trip Logic	1,2	2 trains	O	SR 3.3.1.5	NA
	3$^{(a)}$, 4$^{(a)}$, 5$^{(a)}$	2 trains	C	SR 3.3.1.5	NA

(a) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.

(d) Below the P-6 (Intermediate Range Neutron Flux) interlocks.

(h) Including any reactor trip bypass breakers that are racked in and closed for bypassing an RTB.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.13

SR 3.3.1.13 is the performance of a COT of RPS interlocks every 24 months. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL OPERATIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

The Frequency is based on the known reliability of the interlocks and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

SR 3.3.1.14

SR 3.3.1.14 is the performance of a TADOT of the Manual Reactor Trip, RCP Breaker Position, Turbine Trip Low Auto Stop Oil Pressure, and the SI Input from ESFAS. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable TADOT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. This TADOT is performed every 24 months. The test shall independently verify the OPERABILITY of the tested functions including overlap with the undervoltage and shunt trip mechanisms for the Manual Reactor Trip Function for the Reactor Trip Breakers and Reactor Trip Bypass Breakers up to and including matrix contacts of RT-11/RT-12 from both manual trip actuating devices. The Reactor Trip Bypass Breaker test shall include testing of the automatic undervoltage trip and the shunt trip through the trip actuating devices.

, which is also calibrated ~~calibrated~~ under SR 3.3.1.10,

The Frequency is based on the known reliability of the Functions and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

The SR is modified by a Note that excludes verification of setpoints from the TADOT. Except for Turbine Trip Low Auto Stop Oil Pressure, the Functions affected have no setpoints associated with them.

B 3.3 INSTRUMENTATION

B 3.3.2 Engineered Safety Feature Actuation System (ESFAS) Instrumentation

BASES

BACKGROUND

The ESFAS initiates necessary safety systems, based on the values of selected unit parameters, to protect against violating core design limits and the Reactor Coolant System (RCS) pressure boundary, and to mitigate accidents.

The ESFAS instrumentation is segmented into three distinct but interconnected modules as identified below:

- Field transmitters or process sensors and instrumentation: provide a measurable electronic signal based on the physical characteristics of the parameter being measured,

- Signal processing equipment including analog protection system, field contacts, and protection channel sets: provide signal conditioning, bistable setpoint comparison, process algorithm actuation, compatible electrical signal output to protection system devices, and control board/control room/miscellaneous indications, and

- ESFAS Automatic Actuation Logic and Relays: initiates the proper engineered safety feature (ESF) actuation in accordance with the defined logic and based on the bistable outputs from the signal process control and protection system.

The Allowable Value in conjunction with the trip setpoint and LCO establishes the threshold for ESFAS action to prevent exceeding acceptable limits such that the consequences of Design Basis Accidents (DBAs) will be acceptable. The Allowable Value is considered a limiting value such that a channel is OPERABLE if the setpoint is found not to exceed the Allowable Value during the CHANNEL OPERATIONAL TEST (COT). Note that, although a channel is "OPERABLE" under these circumstances, the ESFAS setpoint must be left adjusted to within the established calibration tolerance band of the ESFAS setpoint in accordance with the uncertainty assumptions stated in the referenced setpoint methodology, (as-left criteria) and confirmed to be operating within the statistical allowances of the uncertainty terms assigned.

add sentence from 3.3.1

BASES

BACKGROUND (continued)

Add same sentence to 3.3.2

assumptions stated in the referenced setpoint methodology (as-left criteria), and confirmed to be operating within the statistical allowances of the uncertainty terms assigned. If the actual setting of the device is found to have exceeded the as-found allowance, the channel would be evaluated to determine Technical Specification OPERABILITY. The results of this evaluation would result in corrective action including those actions required by 10 CFR 50.36 when automatic protective devices do not function as required.

Allowable Values for each RPS function are listed in Table 3.3.1-1. Trip Setpoints that ensure that the Allowable Values are not exceeded over the calibration interval are controlled administratively outside of the Technical Specifications.

During AOOs, which are those events expected to occur one or more times during the unit life, the acceptable limits are:

1. The Departure from Nucleate Boiling Ratio (DNBR) shall be maintained above the Safety Limit (SL) value to prevent departure from nucleate boiling (DNB),

2. Fuel centerline melt shall not occur, and

3. The RCS pressure SL of 2735 psig shall not be exceeded.

Operation within the SLs of Specification 2.0, "Safety Limits (SLs)," also maintains the above values and assures that offsite dose will be within the 10 CFR 50.67 criteria during AOOs.

Accidents are events that are analyzed even though they are not expected to occur during the unit life. The acceptable limit during accidents is that offsite dose shall be maintained within 10 CFR 50.67 limits. 10 CFR 50.67 limits are used in the evaluation of proposed design basis changes with respect to potential reactor accidents of exceedingly low probability of occurrence and low risk of public exposure to radiation. Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event.

The RPS instrumentation is segmented into four distinct but interconnected modules as identified below:

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.5.1	Perform CHANNEL CHECK of the 480 V bus Degraded Voltage Function.	12 hours
SR 3.3.5.2	Perform TADOT of the 480 V bus Degraded Voltage Function.	31 days
SR 3.3.5.3	Perform TADOT of each of the following: a. 480 V bus Undervoltage Function; and b. 480 V bus SBO Function.	24 months
SR 3.3.5.4	Perform ACTUATION LOGIC TEST of each of the following: a. 480 V bus Undervoltage Function; and b. 480 V bus SBO Function.	24 months
SR 3.3.5.5	Perform CHANNEL CALIBRATION with Allowable Values as follows: a. 480 V bus Undervoltage Function Allowable Value: ≥ 206.6 V with a time delay ≤ 3.7 seconds. b. 480 V bus Degraded Voltage Function Allowable Value: ≥ 419 V and ≤ 423 V with time delays as follows: i. ≤ 207 seconds (No SI Signal); and ii. ≥ 8.4 seconds and ≤ 11.6 seconds (Coincident SI). c. 480 V bus SBO Function Allowable Value: ≥ 198.6 V.	24 months ≥ 153 seconds and 11.4

BASES

APPLICABLE SAFETY ANALYSES (continued)

The MSSVs are assumed to have two active and one passive failure modes. The active failure modes are spurious opening, and failure to reclose once opened. The passive failure mode is failure to open upon demand.

The MSSVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

100.6%

The accident analysis requires that five MSSVs per steam generator be OPERABLE to provide overpressure protection for design basis transients occurring at 102% RTP. The LCO requires that five MSSVs per steam generator be OPERABLE in compliance with Reference 2, and the DBA analysis.

The OPERABILITY of the MSSVs is defined as the ability to open upon demand within the setpoint tolerances, to relieve steam generator overpressure, and reseat when pressure has been reduced. The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the Inservice Testing Program.

This LCO provides assurance that the MSSVs will perform their designed safety functions to mitigate the consequences of accidents that could result in a challenge to the RCPB, or Main Steam System integrity.

APPLICABILITY

In MODES 1, 2, and 3, five MSSVs per steam generator are required to be OPERABLE to prevent Main Steam System overpressurization.

In MODES 4 and 5, there are no credible transients requiring the MSSVs. The steam generators are not normally used for heat removal in MODES 5 and 6, and thus cannot be overpressurized; there is no requirement for the MSSVs to be OPERABLE in these MODES.

ACTIONS

The ACTIONS Table is modified by a Note indicating that separate Condition entry is allowed for each MSSV.

With one or more MSSVs inoperable, action must be taken so that the available MSSV relieving capacity meets Reference 2 requirements.

BASES

APPLICABLE SAFETY ANALYSES

The CST provides cooling water to remove decay heat. The minimum amount of water in the condensate storage tank is the amount needed to maintain the plant for 24 hours in MODE 3 following a trip from full power. When the condensate storage tank supply is exhausted, city water will be used.

The CST satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

LCO

100.6%

To satisfy accident analysis assumptions, the CST must contain sufficient cooling water to remove decay heat while in MODE 3 for 24 hours following a reactor trip from 102% RTP. In doing this, it must retain sufficient water to ensure adequate net positive suction head for the AFW pumps during cooldown, as well as account for any losses from the steam driven AFW pump turbine.

The CST required minimum volume of 360,000 gallons includes conservative allowances for instrument accuracy and the unusable volume. When the condensate storage tank supply is exhausted, city water will be used to supply the AFW pumps.

The OPERABILITY of the CST is determined by maintaining the tank level at or above the minimum required level.

APPLICABILITY

In MODES 1, 2, and 3, and in MODE 4, when steam generator is being relied upon for heat removal, the CST is required to be OPERABLE.

In MODE 5 or 6, the CST is not required because the AFW System is not required.

ACTIONS

A.1 and A.2

If the CST is not OPERABLE, the OPERABILITY of the backup supply (City Water) should be verified by administrative means within 4 hours and once every 12 hours thereafter. OPERABILITY of the backup feedwater supply must include verification that City Water is OPERABLE in accordance with requirements specified in the Technical Requirements Manual (TRM) (Ref. 2). The CST must be restored to OPERABLE status within 7 days. The Completion Time for verification of the OPERABILITY of the backup water supply ensures that Condition B is entered promptly if both

ATTACHMENT 2 TO NL-03-154

Clean Typed Pages Affected by Supplement 8

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Table 3.3.1-1 (page 4 of 6)
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FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
17. Reactor Protection System Interlocks					
a. Intermediate Range Neutron Flux, P-6	$2^{(d)}$	2 trains	Q	SR 3.3.1.11 SR 3.3.1.13	≥ 2.5 E-11 amp
b. Low Power Reactor Trips Block, P-7	1	2 trains	R	SR 3.3.1.11 SR 3.3.1.13	NA
c. Power Range Neutron Flux, P-8	1	4	R	SR 3.3.1.11 SR 3.3.1.13	≤ 26.4% RTP
d. Power Range Neutron Flux, P-10	1,2	4	Q	SR 3.3.1.11 SR 3.3.1.13	≤ 10% RTP (set) ≥ 3.6% RTP (reset)
e. Turbine First Stage Pressure, P-7 input	1	2	R	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≤ 9.25% turbine power
18. Reactor Trip Breakers (RTBs)[h]	1,2	2 trains	P	SR 3.3.1.4	NA
	$3^{(a)}, 4^{(a)}, 5^{(a)}$	2 trains	C	SR 3.3.1.4	NA
19. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms	1,2	1 each per RTB	S	SR 3.3.1.4	NA
	$3^{(a)}, 4^{(a)}, 5^{(a)}$	1 each per RTB	C	SR 3.3.1.4	NA
20. Automatic Trip Logic	1,2	2 trains	O	SR 3.3.1.5	NA
	$3^{(a)}, 4^{(a)}, 5^{(a)}$	2 trains	C	SR 3.3.1.5	NA

(a) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.

(d) Below the P-6 (Intermediate Range Neutron Flux) interlocks.

(h) Including any reactor trip bypass breakers that are racked in and closed for bypassing an RTB.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.13

SR 3.3.1.13 is the performance of a COT of RPS interlocks every 24 months. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL OPERATIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

The Frequency is based on the known reliability of the interlocks and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

SR 3.3.1.14

SR 3.3.1.14 is the performance of a TADOT of the Manual Reactor Trip, RCP Breaker Position, Turbine Trip Low Auto Stop Oil Pressure, and the SI Input from ESFAS. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable TADOT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. This TADOT is performed every 24 months. The test shall independently verify the OPERABILITY of the tested functions including overlap with the undervoltage and shunt trip mechanisms for the Manual Reactor Trip Function for the Reactor Trip Breakers and Reactor Trip Bypass Breakers up to and including matrix contacts of RT-11/RT-12 from both manual trip actuating devices. The Reactor Trip Bypass Breaker test shall include testing of the automatic undervoltage trip and the shunt trip through the trip actuating devices.

The Frequency is based on the known reliability of the Functions and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

The SR is modified by a Note that excludes verification of setpoints from the TADOT. Except for Turbine Trip Low Auto Stop Oil Pressure, which is also calibrated under SR 3.3.1.10, the Functions affected have no setpoints associated with them.

B 3.3 INSTRUMENTATION

B 3.3.2 Engineered Safety Feature Actuation System (ESFAS) Instrumentation

BASES

BACKGROUND

The ESFAS initiates necessary safety systems, based on the values of selected unit parameters, to protect against violating core design limits and the Reactor Coolant System (RCS) pressure boundary, and to mitigate accidents.

The ESFAS instrumentation is segmented into three distinct but interconnected modules as identified below:

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- Signal processing equipment including analog protection system, field contacts, and protection channel sets: provide signal conditioning, bistable setpoint comparison, process algorithm actuation, compatible electrical signal output to protection system devices, and control board/control room/miscellaneous indications, and

- ESFAS Automatic Actuation Logic and Relays: initiates the proper engineered safety feature (ESF) actuation in accordance with the defined logic and based on the bistable outputs from the signal process control and protection system.

The Allowable Value in conjunction with the trip setpoint and LCO establishes the threshold for ESFAS action to prevent exceeding acceptable limits such that the consequences of Design Basis Accidents (DBAs) will be acceptable. The Allowable Value is considered a limiting value such that a channel is OPERABLE if the setpoint is found not to exceed the Allowable Value during the CHANNEL OPERATIONAL TEST (COT). Note that, although a channel is "OPERABLE" under these circumstances, the ESFAS setpoint must be left adjusted to within the established calibration tolerance band of the ESFAS setpoint in accordance with the uncertainty assumptions stated in the referenced setpoint methodology, (as-left criteria) and confirmed to be operating within the statistical allowances of the uncertainty terms assigned. If the actual setting of the device is found to have exceeded the as-found allowance, the channel would be evaluated to determine Technical Specification OPERABILITY. The results of this evaluation would result in corrective action including those actions required by 10 CFR 50.36 when automatic protective devices do not function as required.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
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SR 3.3.5.4	Perform ACTUATION LOGIC TEST of each of the following: a. 480 V bus Undervoltage Function; and b. 480 V bus SBO Function.	24 months
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BASES

APPLICABLE SAFETY ANALYSES (continued)

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The MSSVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The accident analysis requires that five MSSVs per steam generator be OPERABLE to provide overpressure protection for design basis transients occurring at 100.6% RTP. The LCO requires that five MSSVs per steam generator be OPERABLE in compliance with Reference 2, and the DBA analysis.

The OPERABILITY of the MSSVs is defined as the ability to open upon demand within the setpoint tolerances, to relieve steam generator overpressure, and reseat when pressure has been reduced. The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the Inservice Testing Program.

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APPLICABILITY

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ACTIONS

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BASES

APPLICABLE SAFETY ANALYSES

The CST provides cooling water to remove decay heat. The minimum amount of water in the condensate storage tank is the amount needed to maintain the plant for 24 hours in MODE 3 following a trip from full power. When the condensate storage tank supply is exhausted, city water will be used.

The CST satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

LCO

To satisfy accident analysis assumptions, the CST must contain sufficient cooling water to remove decay heat while in MODE 3 for 24 hours following a reactor trip from 100.6% RTP. In doing this, it must retain sufficient water to ensure adequate net positive suction head for the AFW pumps during cooldown, as well as account for any losses from the steam driven AFW pump turbine.

The CST required minimum volume of 360,000 gallons includes conservative allowances for instrument accuracy and the unusable volume. When the condensate storage tank supply is exhausted, city water will be used to supply the AFW pumps.

The OPERABILITY of the CST is determined by maintaining the tank level at or above the minimum required level.

APPLICABILITY

In MODES 1, 2, and 3, and in MODE 4, when steam generator is being relied upon for heat removal, the CST is required to be OPERABLE.

In MODE 5 or 6, the CST is not required because the AFW System is not required.

ACTIONS

A.1 and A.2

If the CST is not OPERABLE, the OPERABILITY of the backup supply (City Water) should be verified by administrative means within 4 hours and once every 12 hours thereafter. OPERABILITY of the backup feedwater supply must include verification that City Water is OPERABLE in accordance with requirements specified in the Technical Requirements Manual (TRM) (Ref. 2). The CST must be restored to OPERABLE status within 7 days. The Completion Time for verification of the OPERABILITY of the backup water supply ensures that Condition B is entered promptly if both