Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.7.10:
"Ultimate Heat Sink (UHS)"

PART 1:

Indian Point 3
Improved Technical Specifications and Bases

3.7 PLANT SYSTEMS

3.7.10 Ultimate Heat Sink (UHS)

LCO 3.7.10 The UHS shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. UHS temperature > 95°F. OR UHS inoperable for reasons other than temperature > 95°F.	A.1 Be in MODE 3. AND A.2 Be in MODE 5.	7 hours 37 hours

NRC

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.10.1	Verify average water temperature of UHS is ≤ 95°F.	24 hours

B 3.7 PLANT SYSTEMS

B 3.7.10 Ultimate Heat Sink (UHS)

BASES

BACKGROUND

The UHS provides a heat sink for processing and operating heat from safety related components during a transient or accident, as well as during normal operation. This is done by utilizing the Service Water System (SWS) and the Component Cooling Water (CCW) System.

The ultimate heat sink for IP3 is the Hudson River. The UHS and supporting structures are capable of providing sufficient cooling for thirty days and are sufficient to:

(a) Support simultaneous safe shutdown and cooldown of both operating nuclear units at the Indian Point site and maintain them in a safe condition, and

(b) In the event of an accident in one unit, support required response to that accident and permit simultaneous safe shutdown and cooldown of the remaining unit and maintain them in a safe shutdown condition.

The ultimate heat sink is capable of withstanding the effects of the most severe natural phenomena associated with the Indian Point site, other site related events and a single failure of man-made structural features.

The two principal functions of the UHS are the dissipation of residual heat after reactor shutdown, and dissipation of residual heat after an accident.

APPLICABLE SAFETY ANALYSES

The UHS is the sink for heat removed from the reactor core following all accidents and anticipated operational occurrences in which the unit is cooled down and placed on residual heat removal (RHR) operation. Because IP3 uses the UHS

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

as the normal heat sink for condenser cooling via the Circulating Water System, unit operation at full power is its maximum heat load. Its maximum post accident heat load occurs shortly after a design basis loss of coolant accident (LOCA). Near this time, the unit switches from injection to recirculation and the containment cooling systems and containment recirculation system are required to remove the core decay heat.

The operating limits are based on conservative heat transfer analyses for the worst case LOCA. Reference 1 provides the details of the assumptions used in the analysis, which include worst expected meteorological conditions, conservative uncertainties when calculating decay heat, and worst case single active failure (e.g., single failure of a manmade structure). The UHS meets Regulatory Guide 1.27 (Ref.3), which requires a 30 day supply of cooling water in the UHS.

The UHS satisfies Criterion 3 of 10 CFR 50.36.

LCO

The UHS is required to be OPERABLE and is considered OPERABLE if it contains water at or below the maximum temperature that would allow the SWS to operate for at least 30 days following the design basis LOCA without the loss of net positive suction head (NPSH), and without exceeding the maximum design temperature of the equipment served by the SWS. To meet this condition, the UHS temperature must not exceed 95°F.

APPLICABILITY

In MODES 1, 2, 3, and 4, the UHS is required to support the OPERABILITY of the equipment serviced by the UHS and required to be OPERABLE in these MODES.

In MODE 5 or 6, the OPERABILITY requirements of the UHS are determined by the systems it supports.

(continued)

BASES

ACTIONS

A.1 and A.2

If UHS temperature > 95°F, or is inoperable for reasons other than high temperature, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 7 hours and in MODE 5 within 37 hours.

NRC

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1

This SR verifies that the SWS is available to cool the CCW System to at least its maximum design temperature with the maximum accident or normal design heat loads for 30 days following a Design Basis Accident. The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES. This SR verifies that the average water temperature of the UHS is ≤ 95°F. Requirements for UHS monitoring instrumentation are governed by the Technical Requirements Manual (Ref. 4).

NYPA

REFERENCES

1. FSAR, Section 9.6.
2. WCAP-12313, "Safety Evaluation For An Ultimate Heat Sink Temperature Increase To 95°F At Indian Point Unit 3"
3. Regulatory Guide 1.27.
4. IP3 Technical Requirements Manual.

NYPA

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.7.10:
"ULTIMATE HEAT SINK"

PART 2:

CURRENT TECHNICAL SPECIFICATION PAGES

Annotated to show differences between CTS and ITS

CTS PAGE	AMENDMENT FOR REV O SUBMITTAL	AMENDMENT FOR REV 1 SUBMITTAL	COMMENT
3.3-10	145	145	
3.3-10a	98	98	
3.3-19	145;97-175	145;9-22-98	Change to Bases Page. No impact to ITS 3.7.10
T4.1-1(5)	169;98-043	185	No impact
T4.1-1(6)	181;98-043	185	No impact

SEE ITS 3.7.8

3. If the Component Cooling System is not restored to meet the requirements of 3.3.E.1 within the time periods specified in 3.3.E.2, then:

   a. If the reactor is critical, it shall be in the hot shutdown condition within four hours and in the cold shutdown condition within the following 24 hours.

   b. If the reactor is subcritical, the reactor coolant system temperature and pressure shall not be increased more than 25°F and 100 psi, respectively, over existing values. If the requirements of 3.3.E.1 are not satisfied within an additional 48 hours, the reactor shall be brought to the cold shutdown condition utilizing normal operating procedures. The shutdown shall start no later than the end of the 48 hour period.

SEE ITS 3.7.9

LCO 3.7.10, Applicability

~~F.~~ Service Water System/Ultimate Heat Sink

~~The reactor shall not be brought above cold shutdown unless~~: (Mode 1, 2, 3 and 4) (A3)

SEE ITS 3.7.9

   a. Three service water pumps on the designated essential header and a minimum of two service water pumps on the designated non-essential header, together with their associated piping and valves, are operable.

LCO 3.7.10, SR 3.7.10.1

   b. (Verify every 24 hours that) The service water inlet temperature is less than or equal to 95°F. (M.1)

SEE ITS 3.7.9

2. When the reactor is above cold shutdown and if the requirements of 3.3.F.1.a cannot be met within twelve hours, the reactor shall be brought to the cold shutdown condition, starting no later than the end of the twelve hour period, utilizing normal operating procedures.

LCO 3.7.10 ~~3.~~ Cond A, ~~Req. Act B.1, Req. Act B.2~~

When the reactor is above cold shutdown and if the requirement of 3.3.F.1.b is exceeded, the reactor shall be placed in at least hot shutdown within seven (6 hours) hours, and in cold shutdown within ~~the following thirty~~ (36) hours unless the service water inlet temperature decreases to within the requirement of 3.3.F.1.b. (A.4)

~~(Add Condition A and associated Req. Act) (A.5)~~

R.1

Amendment No. ~~74~~, ~~98~~, 145

SEE ITS 3.7.9

4. Isolation shall be maintained between the essential and non-essential headers at all times when above cold shutdown conditions except that for a period of eight hours the headers may be connected while another essential header is being placed in service as described in F.2 above.

5. At least two service water inlet temperature monitoring instruments (any combination of installed or portable instruments) shall be operable when the reactor is above 350°F and service water inlet temperature exceeds 90°F.

6. If the requirements of 3.3.F.5 cannot be met, the reactor shall be placed in the hot shutdown condition within the next seven hours and subsequently cooled below 350°F using normal operating procedures.

7. Service water inlet temperature shall be the average of two or more service water inlet temperature monitoring instrument readings per 3.3.F.5 taken within a five minute interval (instantaneous).

8. When the reactor is above 350°F and service water inlet temperature per 3.3.F.7 exceeds 90°F, service water inlet temperature monitoring shall commence at a frequency of once per hour.

LA.1

ITS 3.7/10 rev. 1

A total of six service water pumps are installed. Only two of the set of three service water pumps on the header designated the essential header are required immediately following a postulated loss-of-coolant accident.[8] During the recirculation phase of the accident, two service water pumps on the non-essential header will be manually started to supply cooling water for one component cooling system heat exchanger, one control room air conditioner, and one diesel generator; the other component cooling system heat exchanger, the other control room air conditioner, the two other diesel generators and remaining safety related equipment are cooled by the essential service water header.[14] During the recirculation phase of the acccident, both control room air conditioner units may be cooled by the essential service water header.

The operability requirements on service water temperature monitoring instrumentation and the frequency of service water temperature monitoring insures that appropriate action can be taken to preclude operation beyond established limits. The locations selected for monitoring river water temperature are typically at the circulating or service water inlets, at the circulating water inlet boxes to the condenser hotwells or at the service water supply header to the fan cooler units. Temperature measurements at each of these locations are representative of the river water temperature supplied to cool plant heat loads. Alternate locations may be acceptable on this basis. The limit on the service water maximum inlet temperature insures that the service water and component cooling water systems will be able to dissipate the heat loads generated in the limiting design basis accident[15]. This restriction allows up to seven hours for river water temperature transients which may temporarily increase the service water inlet temperature due to tidal effects to dissipate.

The operability of the equipment and systems required for the control of hydrogen gas ensures that this equipment is available to maintain the hydrogen concentration within containment below the flammable limit during post-LOCA conditions. Hydrogen concentration exceeding the flammable limit could potentially result in a containment wide hydrogen burn. This could lead to overpressurization of containment, a breach of CONTAINMENT INTEGRITY, containment leakage, unacceptably high offsite doses, and damage to safety-related equipment located in containment. Two full rated recombiner units are provided in order to control the hydrogen evolved in containment following a loss-of-coolant accident. Each unit is capable of preventing the hydrogen concentration from exceeding the flammable limit. Each recombiner is installed such that independence is maintained and redundancy is assured. Each hydrogen recombiner system consists of a recombiner located inside containment, and a separate power supply, and control panel located outside containment such that they are accessible following a design basis accident.

A.1

SEE CTS MASTER MARKUP

TABLE 4.1-1 (Sheet 5 of 6)

Channel Description	Check	Calibrate	Test	Remarks
37. Core Exit Thermocouples	D	24M	N.A.	
38. Overpressure Protection System (OPS)	D	18M (1)	18M	1) Calibration frequency for OPS sensors (RCS pressure and temperature) is 24 months
39. Reactor Trip Breakers	N.A.	N.A.	TM(1)	1) Independent operation of under-voltage and shunt trip attachments
			24M(2)	2) Independent operation of under-voltage and shunt trip from Control Room manual push-button
40. Reactor Trip Bypass Breakers	N.A.	N.A.	(1)	1) Manual shunt trip prior to each use
			24M(2)	2) Independent operation of under-voltage and shunt trip from Control Room manual push-button
			24M(3)	3) Automatic undervoltage trip
41. Reactor Vessel Level Indication System (RVLIS)	D	24M	N.A.	
42. Ambient Temperature Sensors Within the Containment Building	D	24M	N.A.	
43. River Water Temperature # (installed)	S	18M	N.A.	1) Check against installed instrumentation or another portable device
44. River Water Temperature # (portable)	S (1)	Q (2)	N.A.	2) Calibrate within 30 days prior to use and quarterly thereafter
45. Steam Line Flow	S	24M	Q	Engineered Safety Features circuits only

SR 3.7.10.1

LA.1

ITS 3.7.10

Amendment No. ~~28~~, ~~34~~, ~~65~~, ~~74~~, ~~78~~, ~~93~~, ~~98~~, ~~107~~, ~~125~~, ~~126~~, ~~137~~, ~~140~~, ~~142~~, ~~164~~, ~~169~~, TSCR 98-043

Superceded by Amendment 185
No impact on ITS 3.7.10. See Next page

TABLE 4.1-1 (Sheet 5 of 6)

Channel Description	Check	Calibrate	Test	Remarks
37. Core Exit Thermocouples	D	24M	N.A.	
38. Overpressure Protection System (OPS)	D	18M (1)	24M	1) Calibration frequency for OPS sensors (RCS pressure and temperature) is 24 months
39. Reactor Trip Breakers	N.A.	N.A.	TM(1)	1) Independent operation of under-voltage and shunt trip attachments
			24M(2)	2) Independent operation of under-voltage and shunt trip from Control Room manual push-button
40. Reactor Trip Bypass Breakers	N.A.	N.A.	(1)	1) Manual shunt trip prior to each use
			24M(2)	2) Independent operation of under-voltage and shunt trip from Control Room manual push-button
			24M(3)	3) Automatic undervoltage trip
41. Reactor Vessel Level Indication System (RVLIS)	D	24M	N.A.	
42. Ambient Temperature Sensors Within the Containment Building	D	24M	N.A.	
43. River Water Temperature # (installed)	S	18M	N.A.	
44. River Water Temperature # (portable)	S (1)	Q (2)	N.A.	1) Check against installed instrumentation or another portable device
				2) Calibrate within 30 days prior to use and quarterly thereafter
45. Steam Line Flow	S	24M	Q	Engineered Safety Features circuits only

Amendment No. ~~28, 54, 65, 74, 78, 93, 98, 107, 125, 126, 137, 140, 142, 164, 169~~, 185

ITS 3.7.10 rev 1

Table Notation

SEE CTS MASTER MARKUP

*	By means of the movable incore detector system
**	Quarterly when reactor power is below the setpoint and prior to each startup if not done previous month.

SR 3.7.10.1

#	These requirements are applicable when specification 3.3.F.5 is in effect only.

LA.1

SEE CTS MASTER MARKUP

##	The "each shift" frequency also requires verification that the DNB parameters (Reactor Coolant Temperature, Reactor Coolant Flow, and Pressurizer Pressure) are within the limits of Technical Specification 3.1.H.

- S - Each Shift
- W - Weekly
- P - Prior to each startup if not done previous week
- M - Monthly
- NA - Not Applicable
- Q - Quarterly
- D - Daily
- 18M - At least once per 18 months
- TM - At least every two months on a staggered test basis (i.e., one train per month)
- 24M - At least once per 24 months
- 6M - At least once per 6 months

Superceded by Amend 185
No impact on ITS 3.7.10
See next page

TABLE 4.1-1 (Sheet 6 of 6)

Table Notation

*	By means of the movable incore detector system
**	Quarterly when reactor power is below the setpoint and prior to each startup if not done previous month.
#	These requirements are applicable when specification 3.3.F.5 is in effect only.
##	The "each shift" frequency also requires verification that the DNB parameters (Reactor Coolant Temperature, Reactor Coolant Flow, and Pressurizer Pressure) are within the limits of Technical Specification 3.1.H.

S - Each Shift (i.e.,at least once per 12 hours)
W - Weekly
P - Prior to each startup if not done previous week
M - Monthly
NA - Not Applicable
Q - Quarterly
D - Daily
18M - At least once per 18 months
TM - At least every two months on a staggered test basis (i.e., one train per month)
24M - At least once per 24 months
6M - At least once per 6 months

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.7.10: "Ultimate Heat Sink (UHS)"

PART 3:

DISCUSSION OF CHANGES

Differences between CTS and ITS

DISCUSSION OF CHANGES
ITS SECTION 3.7.10 - Ultimate Heat Sink (UHS)

ADMINISTRATIVE

A.1 In the conversion of the Indian Point Unit 3 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Additionally, editorial changes, reformatting, and revised numbering are adopted to make ITS consistent with the conventions in NUREG-1431, Standard Technical Specifications, Westinghouse Plants, Rev. 1, i.e., the Improved Standard Technical Specifications.

The CTS Bases are deleted and replaced with comprehensive ITS Bases designed to support interpretation and implementation of the associated Technical Specifications. The Bases explain, clarify, and document the reasons (i.e., bases) for the associated Technical Specifications, and reflect the IP3 plant specific design, analyses, and licensing basis. In accordance with 10 CFR 50.36(a), the ITS Bases are included with the proposed ITS conversion application; however, deletion of the CTS Bases and the adoption of the ITS Bases is an administrative change with no impact on safety.

A.2 CTS Limiting Conditions for Operation (LCOs) and Surveillance Requirements (SRs) include statements of the objective and the applicability. The CTS statements of objective and applicability are deleted because these statements do not establish any requirements and do not provide any guidance for the application of CTS requirements. Therefore, deletion of these statements has no significant adverse impact on safety.

A.3 CTS 3.3.F specifies the Applicability for Service Water System/Ultimate Heat Sink as whenever the reactor is above cold shutdown (i.e., Modes 1, 2, 3 and 4). ITS 3.7.9 and ITS LCO 3.7.10 maintain this Applicability by requiring that the Service Water System and Ultimate Heat Sink be Operable in Modes 1, 2, 3 and 4. This is an administrative change with no impact on safety because there is no change to the CTS Applicability.

A.4 CTS 3.3.F.3 requires that the reactor be placed in at least hot shutdown (i.e., Mode 3) within 7 hours, and in cold shutdown (i.e., Mode 5) within the following 30 hours, if the service water inlet temperature exceeds 95°F. Under the same condition, ITS LCO 3.7.10 Required Actions B.1 and B.2 require that the reactor be placed in Mode 3 within 6 hours and in Mode 5 within 36 hours. This change is needed because it makes the plant shutdown Completion Times consistent with other ITS LCOs. This change is acceptable because the ITS Completion Times allow sufficient time to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. This is an administrative change with no impact on safety because there is no significant change to the existing requirements.

A.5 CTS 3.3.F.1.b requires that UHS (service water inlet) temperature is less than 95°F. The CTS Bases for CTS 3.3.F.1.b clarifies that this restriction is intended to allow up to seven hours for the dissipation of tidal effects that can cause river water temperature transients that may temporarily increase UHS temperature.

ITS LCO 3.7.10, Condition A and Required Action A.1, maintain the requirement that UHS be maintained less than 95°F. Additionally, the Required Action A.1 Completion Time of 7 hours for restoration of UHS temperature before a reactor shutdown must be initiated provides explicit recognition of the allowance for the dissipation of tidal effects that can cause river water temperature transients that may temporarily increase UHS temperature. This is an administrative change with no adverse impact of safety because it is an explicit statement of an existing Completion Time stated in the CTS Bases.

R.1

MORE RESTRICTIVE

M.1 CTS 3.3.F.1.b requires that UHS (service water inlet) temperature is less than 95°F: however, there is no requirement for periodic verification that this limit is met except that CTS 3.3.F.8 includes a requirement to monitor UHS temperature every hour when the UHS temperature exceeds 90°F. ITS SR 3.7.10.1 requires verification every 24 that the average temperature of the Ultimate Heat Sink is ≤ 95°F regardless of the UHS temperature; and, requirements for accelerated verification of UHS temperature when UHS temperature is approaching the upper limit are relocated to the Technical Requirements Manual (TRM) (See ITS 3.7.10, DOC LA.1). This more restrictive requirement is acceptable because it verifies that the UHS temperature is consistent with the accident analysis assumptions, and the 24 hour Frequency is acceptable based on operating experience related to trending of the parameter variations during the applicable modes. This change has no adverse impact on safety.

LESS RESTRICTIVE

None

REMOVED DETAIL

LA.1 CTS 3.3.F.5 through CTS 3.3.F.8 require accelerated monitoring (once per hour) using specific instruments whenever the UHS temperature is > 90°F (i.e., approaching the LCO limit of 95°F). CTS Table 4.1-1, Items 43 and 44 require periodic channel checks and calibrations of the instruments used to perform these verifications.

ITS LCO 3.7.10 maintains the requirement that UHS be maintained less than 95°F and ITS SR 3.7.10.1 maintains the requirement for verification every 24 hours that this limit is met; however, requirements for accelerated monitoring of UHS temperature using specific instruments whenever the UHS temperature is approaching the LCO limit of 95°F are relocated to the TRM.

This change is acceptable because ITS LCO 3.7.10 maintains the requirement that the LCO is met whenever the plant is in the Applicable Modes. Maintaining this requirement in Technical Specifications and maintaining requirements for accelerated monitoring of UHS temperature using specific instruments whenever the UHS temperature is approaching the LCO limit provides an adequate level of assurance of prompt identification and initiation of actions if UHS temperature exceeds required limits.

The Quality Assurance Plan will be revised to specify that requirements in the TRM are part of the facility as described in the FSAR and that changes to the TRM can be made only in accordance with the requirements of 10 CFR 50.59. Therefore, this change is acceptable because there is no change to the existing requirements by the relocation of requirements to the TRM and future changes to the TRM will be controlled in accordance with 10 CFR 50.59.

This change is a less restrictive administrative change with no impact on safety because ITS 3.7.10 maintains the requirement that plant operation be curtailed if UHS temperature exceeds the upper limit. Therefore, requirements for accelerated monitoring of UHS temperature and the instruments used to perform this accelerated monitoring can be maintained in the TRM with no significant adverse impact on safety.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.7.10: "Ultimate Heat Sink (UHS)"

PART 4:

No Significant Hazards Considerations for Changes between CTS and ITS that are Less Restrictive

No Significant Hazard Considerations for Changes that are Administrative, More Restrictive, and Removed Details are the same for all Packages. A Copy is included at the end of the Package.

NO SIGNIFICANT HAZARDS EVALUATION
ITS SECTION 3.7.10 - Ultimate Heat Sink (UHS)

LESS RESTRICTIVE
("L.1" Labeled Comments/Discussions)

There are no less restrictive changes for the adoption of this ITS.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.7.10:
"Ultimate Heat Sink (UHS)"

PART 5:

NUREG-1431
Annotated to show differences between NUREG-1431 and ITS

Status of NUREG 1431 Generic Changes for ITS 3.7.10

This ITS Specification is based on NUREG-1431 Specification No. **3.7.9**
as modified by the following Generic Changes:

OG No.	TSTF No.	Generic Change Description	NRC STATUS	IP3 STATUS	JD No.
N/A	N/A	NO GENERIC CHANGES ARE POSTED AGAINST THIS SPECIFICATION.	Not Applicable	Not Applicable	N/A

<CTS>

3.7 PLANT SYSTEMS

3.7.9 10 Ultimate Heat Sink (UHS)

<3.3.F> <3.3.F.1.b> LCO 3.7.8 10 The UHS shall be OPERABLE.

<3.3.F.1> <DOC A.3> APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

Insert 3.7-21-01 Insert 3.7-21-02

CONDITION	REQUIRED ACTION	COMPLETION TIME
<DOC A.5> [A. One or more cooling towers with one cooling tower fan inoperable.	A.1 Restore cooling tower fan(s) to OPERABLE status.	7 days 7 hours] CLB.1
<3.3.F.3> A. [B. Required Action and associated Completion Time of Condition A not met. OR] UHS inoperable [for reasons other than Condition A]. temperature > 95°F <3.3.F.3> <DOC A.4>	A B.1 Be in MODE 3. AND A B.2 Be in MODE 5.	7 6 hours CLB.1 37 36 hours

R.1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
[SR 3.7.9.1 Verify water level of UHS is ≥ [562] ft [mean sea level].	[24] hours]

(continued)

NUREG-1431 Markup Inserts
ITS SECTION 3.7.10 - Ultimate Heat Sink (UHS)

INSERT: 3.7-21-01

UHS temperature > 95°F.

INSERT: 3.7-21-02

Verify UHS temperature ≤ 95°F.

SURVEILLANCE REQUIREMENTS (continued)

<3.3.F.1.b>
<DOC M.1>

SURVEILLANCE		FREQUENCY
SR 3.7.9.2 (10)	Verify average water temperature of UHS is ≤ [90]°F. (95)	24 hours
SR 3.7.9.3	Operate each cooling tower fan for ≥ [15] minutes.	31 days
SR 3.7.9.4	Verify each cooling tower fan starts automatically on an actual or simulated actuation signal.	[18] months

B 3.7 PLANT SYSTEMS

B 3.7.9 10 Ultimate Heat Sink (UHS)

BASES

BACKGROUND

The UHS provides a heat sink for processing and operating heat from safety related components during a transient or accident, as well as during normal operation. This is done by utilizing the Service Water System (SWS) and the Component Cooling Water (CCW) System.

Insert: B3.7-46-01

The UHS has been defined as that complex of water sources, including necessary retaining structures (e.g., a pond with its dam, or a river with its dam), and the canals or conduits connecting the sources with, but not including, the cooling water system intake structures as discussed in the FSAR, Section [9.2.5] (Ref. 1). If cooling towers or portions thereof are required to accomplish the UHS safety functions, they should meet the same requirements as the sink. The two principal functions of the UHS are the dissipation of residual heat after reactor shutdown, and dissipation of residual heat after an accident.

A variety of complexes is used to meet the requirements for a UHS. A lake or an ocean may qualify as a single source. If the complex includes a water source contained by a structure, it is likely that a second source will be required.

The basic performance requirements are that a 30 day supply of water be available, and that the design basis temperatures of safety related equipment not be exceeded. Basins of cooling towers generally include less than a 30 day supply of water, typically 7 days or less. A 30 day supply would be dependent on other source(s) and makeup system(s) for replenishing the source in the cooling tower basin. For smaller basin sources, which may be as small as a 1 day supply, the systems for replenishing the basin and the backup source(s) become of sufficient importance that the makeup system itself may be required to meet the same design criteria as an Engineered Safety Feature (e.g., single failure considerations), and multiple makeup water sources may be required.

(continued)

INSERT: B 3.7-46-01

The ultimate heat sink for IP3 is the Hudson River. The UHS and supporting structures are capable of providing sufficient cooling for thirty days and are sufficient to:

DB.1

(a) Support simultaneous safe shutdown and cooldown of both operating nuclear units at the Indian Point site and maintain them in a safe condition, and

(b) In the event of an accident in one unit, support required response to that accident and permit simultaneous safe shutdown and cooldown of the remaining unit and maintain them in a safe shutdown condition.

The ultimate heat sink is capable of withstanding the effects of the most severe natural phenomena associated with the Indian Point site, other site related events and a single failure of man-made structural features.

BASES

BACKGROUND (continued)

Additional information on the design and operation of the system, along with a list of components served, can be found in Reference 1.

APPLICABLE SAFETY ANALYSES

The UHS is the sink for heat removed from the reactor core following all accidents and anticipated operational occurrences in which the unit is cooled down and placed on residual heat removal (RHR) operation. For units that use UHS as the normal heat sink for condenser cooling via the Circulating Water System, unit operation at full power is its maximum heat load. Its maximum post accident heat load occurs 20 minutes after a design basis loss of coolant accident (LOCA). Near this time, the unit switches from injection to recirculation and the containment cooling systems and RHR are required to remove the core decay heat.

Because IP3 uses the

shortly

containment recirculation system

The operating limits are based on conservative heat transfer analyses for the worst case LOCA. Reference 1 provides the details of the assumptions used in the analysis, which include worst expected meteorological conditions, conservative uncertainties when calculating decay heat, and worst case single active failure (e.g., single failure of a manmade structure). The UHS is designed in accordance with Regulatory Guide 1.27 (Ref. 2), which requires a 30 day supply of cooling water in the UHS.

meets

3

The UHS satisfies Criterion 3 of the NRC Policy Statement.

10 CFR 50.36

LCO

The UHS is required to be OPERABLE and is considered OPERABLE if it contains a sufficient volume of water at or below the maximum temperature that would allow the SWS to operate for at least 30 days following the design basis LOCA without the loss of net positive suction head (NPSH), and without exceeding the maximum design temperature of the equipment served by the SWS. To meet this condition, the UHS temperature should not exceed [90°F] and the level should not fall below [562 ft mean sea level] during normal unit operation.

must

95°F

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, 3, and 4, the UHS is required to support the OPERABILITY of the equipment serviced by the UHS and required to be OPERABLE in these MODES.

In MODE 5 or 6, the OPERABILITY requirements of the UHS are determined by the systems it supports.

ACTIONS

A.1

Insert: B3.7-48-01

If one or more cooling towers have one fan inoperable (i.e., up to one fan per cooling tower inoperable), action must be taken to restore the inoperable cooling tower fan(s) to OPERABLE status within 7 days.

The 7 day Completion Time is reasonable based on the low probability of an accident occurring during the 7 days that one cooling tower fan is inoperable (in one or more cooling towers), the number of available systems, and the time required to reasonably complete the Required Action.

B.1 and B.2

Insert: B3.7-48-02

If the cooling tower fan cannot be restored to OPERABLE status within the associated Completion Time, or if the UHS is inoperable for reasons other than Condition A, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

R.1

SURVEILLANCE REQUIREMENTS

SR 3.7.9.1

This SR verifies that adequate long term (30 day) cooling can be maintained. The specified level also ensures that sufficient NPSH is available to operate the SWS pumps. The [24] hour Frequency is based on operating experience related to trending of the parameter variations during the

(continued)

NUREG-1431 Markup Inserts
ITS SECTION 3.7.10 - Ultimate Heat Sink (UHS)

INSERT: B 3.7-48-01

or is inoperable for reasons other than high temperature

If UHS temperature > 95°F, the UHS temperature must be verified to be ≤ 95°F within 7 hours. The 7 hour Completion Time allows for the dissipation of tidal effects that can cause river water temperature transients that may temporarily increase localized UHS temperature.

INSERT: B 3.7-48-02

UHS temperature does not return to ≤ 95°F

R.1

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.9.1 (continued)

applicable MODES. This SR verifies that the UHS water level is ≥ [562] ft [mean sea level].

SR 3.7.9.2 10.1

This SR verifies that the SWS is available to cool the CCW System to at least its maximum design temperature with the maximum accident or normal design heat loads for 30 days following a Design Basis Accident. The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES. This SR verifies that the average water temperature of the UHS is ≤ [90°F]. 95°F

Insert: B 3.7-49-01

R. 1

SR 3.7.9.3

Operating each cooling tower fan for ≥ [15] minutes ensures that all fans are OPERABLE and that all associated controls are functioning properly. It also ensures that fan or motor failure, or excessive vibration, can be detected for corrective action. The 31 day Frequency is based on operating experience, the known reliability of the fan units, the redundancy available, and the low probability of significant degradation of the UHS cooling tower fans occurring between surveillances.

SR 3.7.9.4

This SR verifies that each cooling tower fan starts and operates on an actual or simulated actuation signal. The [18] month Frequency is consistent with the typical refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section [9.2.5]. 9.6
2. 3 Regulatory Guide 1.27.

Insert: B3.7-49-02 Insert: B 3.7-49-03

R 1

NUREG-1431 Markup Inserts
ITS SECTION 3.7.10 - Ultimate Heat Sink (UHS)

INSERT: B 3.7-49-01

Requirements for UHS monitoring instrumentation are governed by the Technical Requirements Manual (Ref. 4) R-1

INSERT: B 3.7-49-02

2. WCAP-12313, "Safety Evaluation For An Ultimate Heat Sink Temperature Increase To 95°F At Indian Point Unit 3"

INSERT: B 3.7-49-03

4. IP3 Technical Requirements Manual. R-1

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.7.10: "Ultimate Heat Sink (UHS)"

PART 6:

Justification of Differences between

NUREG-1431 and IP3 ITS

JUSTIFICATION OF DIFFERENCES FROM NUREG-1431
ITS SECTION 3.7.10 - Ultimate Heat Sink (UHS)

RETENTION OF EXISTING REQUIREMENT (CURRENT LICENSING BASIS)

CLB.1 NUREG-1431, Rev 1, Section 3.7.10, was modified as needed to reflect the IP3 design and current licensing basis. A detailed description of the design, accident analysis assumptions, and Operability requirements are incorporated into the IP3 ITS Bases. These changes maintain the IP3 current licensing basis except as identified and justified in the CTS/ITS discussion of changes.

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT

PA.1 Corrected typographical error or made a minor editorial improvement to improve clarity and ensure requirements are fully understood and consistently applied. There are no technical changes to requirements as specified in NUREG 1431, Rev. 1; therefore, this change is not a significant or generic deviation from NUREG 1431, Rev 1.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN OR DESIGN BASIS

DB.1 Design or implementation details are incorporated or revised as necessary to more precisely describe IP3 current design or practice. These changes are intended to describe the design, improve clarity, or ensure requirements are fully understood and consistently applied. Unless identified and described below, these changes are self-explanatory. A detailed description of the design, accident analysis assumptions, and Operability requirements are incorporated into the IP3 ITS Bases. These changes maintain the IP3 current licensing basis except as identified and justified in the CTS/ITS discussion of changes. There are no technical changes to requirements as specified in NUREG 1431, Rev 1; therefore, this change is not a significant or generic deviation from NUREG 1431, Rev 1.

DIFFERENCE BASED ON A GENERIC CHANGE TRAVELER FOR NUREG-1431

None

JUSTIFICATION OF DIFFERENCES FROM NUREG-1431
ITS SECTION 3.7.10 - Ultimate Heat Sink (UHS)

DIFFERENCE FOR ANY REASON OTHER THAN ABOVE

None

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.7.11:
"Control Room Ventilation System (CRVS)"

PART 1:

Indian Point 3
Improved Technical Specifications and Bases

No changes to Rev 0 Spec. pages

B 3.7 PLANT SYSTEMS

B 3.7.11 Control Room Ventilation System (CRVS)

BASES

BACKGROUND

The CRVS provides a protected environment from which operators can control the unit following an uncontrolled release of radioactivity, chemicals, or toxic gas.

The Control Room Ventilation System consists of the following equipment: a single filter unit consisting of two roughing filters, two high efficiency particulate air (HEPA) filters; two activated charcoal adsorbers for removal of gaseous activity (principally iodines); two 100% capacity filter booster fans; and, a single duct system including dampers, controls and associated accessories to provide for three different air flow configurations. The air-conditioning units associated with the CRVS are governed by LCO 3.7.12, "Control Room Air Conditioning System (CRACS)."

The CRVS is divided into two trains with each train consisting of a filter booster fan and the associated inlet damper and the following components which are common to both trains: the control room filter unit, damper A (filter unit bypass for outside air makeup to the Control Room), damper B (filter unit inlet for outside air makeup to the Control Room), damper C (filter unit inlet for reticulated air), and the toilet and locker room exhaust fan. The two filter booster fans (F 31 and F 32) are powered from safeguards power trains 5A (EDG 33) and 6A (EDG 32), respectively. The automatic dampers that are common to both trains are positioned in the fail-safe position (open or closed) by either of the redundant actuation channels.

The CRVS is an emergency system, parts of which operate during normal unit operations.

The three different CRVS air flow configurations are as follows:

(continued)

BASES

BACKGROUND (continued)

a) Normal operation consists of approximately 85% (8500 cfm) unfiltered recirculated flow driven by the air-conditioning fans and approximately 15% (1500 cfm) unfiltered outside air makeup;

b) Incident mode with outside air makeup (i.e. 10% incident mode) consists of approximately 87% (9250 cfm) unfiltered recirculated flow driven by the two safety related air-conditioning fans, at least 10% (> 1000 cfm) filtered recirculated flow driven by either one of the two filter booster fans and approximately 2.5% to 4.0% (250 to 400 cfm) filtered outside air makeup;

c) Incident mode with no outside air makeup (i.e. 100% incident mode) consists of 85% (9100 cfm) unfiltered recirculated flow driven by the two safety related air-conditioning fans, approximately 15% filtered recirculated flow driven by either one of the two filter booster fans and no outside air makeup. RAI-04

Note that the required recirculation rates are demonstrated with surveillance tests conducted with the air conditioning system (CRACS) operating. An inoperable CRACS fan will affect the flow balance of the CRVS due to interconnected ductwork. Therefore, if the fan associated with one of the air-conditioning units governed by LCO 3.7.12 is inoperable, Conditions in both LCO 3.7.11, Control Room Ventilation System, and LCO 3.7.12, Control Room Air Conditioning System (CRACS), will apply. RAI-04

Incident mode with outside air makeup is the preferred method of operation during any radiological event because it provides outside air for pressurization of the Control Room. Calculations indicate that very low volumes of outside air makeup will maintain the Control Room at a slight positive pressure. Nevertheless, due to the difficulty of adjusting and maintaining the flow dampers to provide a low flow, it was determined that the damper should be adjusted to provide a flow of approximately 250 cfm (2.5% outside air makeup). However, a higher volume of outside air makeup to

(continued)

BASES

BACKGROUND (continued)

the Control Room increase the thyroid dose to the operators during an accident. Therefore, the Control Room dose assessment assumes a filtered outside air makeup of approximately 400 cfm (4.0% outside air makeup).

On a Safety Injection signal or high radiation in the Control Room (Radiation Monitor R-1), the CRVS will actuate to the incident mode with outside air makeup (i.e. 10% incident mode). This will cause one of the two filters booster fans to start, the locker room exhaust fan to stop, and CRVS dampers to open or close as necessary to filter all incoming outside air and direct approximately 10% of the recirculated air through the filter unit. In the event that the first booster fan fails to start, the second booster fan will start after a predetermined time delay.

If for any reason it is required or desired to operate with 100% recirculated air (e.g., toxic gas condition is identified), the CRVS can be placed in the incident mode with no outside air makeup (i.e. 100% incident mode) by remote manually operated switches. The Firestat detectors will also initiate 100% incident mode in the CRVS.

The control room is continuously monitored by radiation and toxic gas detectors. On a Safety Injection signal or high radiation in the Control Room (Radiation Monitor R-1), will cause actuation of the emergency radiation state of the CRVS (i.e., incident mode with outside air makeup (i.e. 10% incident mode)).

The CRVS does not actuate automatically in response to toxic gases. Separate chlorine, ammonia and oxygen probes are provided to detect the presence of these gases in the outside air intake. Additionally, monitors in the Control Room will detect low oxygen levels and high levels of chlorine and ammonia. The CRVS may be placed in the incident mode with no outside air makeup (i.e. 100% incident mode) to respond to these conditions. Instrumentation for toxic gas monitoring is governed by the IP3 Technical Requirements Manual (TRM) (Ref. 4). Generally, the manually initiated actions of the toxic gas isolation state are more restrictive, and will override the actions of the emergency radiation state.

(continued)

BASES

BACKGROUND (continued)

A single train will create a slight positive pressure in the control room. The CRVS operation in maintaining the control room habitable is discussed in the FSAR, Section 9.9 (Ref. 1).

The CRVS is designed in accordance with Seismic Category I requirements.

The CRVS is designed to maintain the control room environment for 30 days of continuous occupancy after a Design Basis Accident (DBA) without exceeding a 5 rem whole body dose or 30 rem to the thyroid.

APPLICABLE SAFETY ANALYSES

The CRVS active components are arranged in redundant, safety related ventilation trains. The location of components and ducting within the control building envelope ensures an adequate supply of filtered air to all areas requiring access. The CRVS provides airborne radiological protection for the control room operators, as demonstrated by the control room accident dose analyses for the most limiting design basis accident (i.e., DBA LOCA) fission product release presented in the FSAR, Chapter 14 (Ref. 2).

NYPA

Radiation monitor R-1 is not required for the Operability of the Control Room Ventilation System because control room isolation is initiated by the safety injection signal in MODES 1, 2, 3, 4, and control room isolation is not required for maintaining radiation exposure within General Design Criteria 19 limits following a fuel handling accident or gas-decay-tank rupture.

The worst case active failure of a component of the CRVS, assuming a loss of offsite power, does not impair the ability of the system to perform its design function. However, the original CRVS design was not required to meet single failure criteria and, although upgraded from the original design, CRVS does not satisfy all requirements in IEEE-279 for single failure tolerance.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Each of the automatic dampers that are common to both trains is positioned in the fail-safe position (open or closed) by either of the redundant actuation channels.

NYPA

The CRVS satisfies Criterion 3 of 10 CFR 50.36.

LCO

Two CRVS trains are required to be OPERABLE to ensure that at least one is available. Total system failure could result in exceeding a dose of 5 rem whole body or 30 rem to the thyroid of the control room operator in the event of a large radioactive release.

The CRVS is considered OPERABLE when the individual components necessary to limit operator exposure are OPERABLE in both trains. A CRVS train is OPERABLE when the associated:

a. Filter booster fan and an air-conditioning unit fan powered from the same safeguards power train are OPERABLE;

b. HEPA filters and charcoal adsorbers are not excessively restricting flow, and are capable of performing their filtration functions; and

c. Ductwork, valves, and dampers are OPERABLE or in the incident mode, and air circulation can be maintained.

In addition, the control room boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors.

Instrumentation for toxic gas monitoring is governed by the IP3 Technical Requirements Manual (TRM) (Ref. 4) and is not included in the LCO.

Note that the required recirculation rates are demonstrated with surveillance tests conducted with the air conditioning system (CRACS) operating. An inoperable CRACS fan will affect the flow

(continued)

BASES

LCO (continued)	balance of the CRVS due to interconnected ductwork. Therefore, if the fan associated with one of the air-conditioning units governed by LCO 3.7.12 is inoperable, Conditions in both LCO 3.7.11, Control Room Ventilation System, and LCO 3.7.12, Control Room Air Conditioning System (CRACS), will apply.
APPLICABILITY	In MODES 1, 2, 3, 4 CRVS must be OPERABLE to limit operator exposure during and following a DBA. The CRVS is not required in MODE 5 or 6, or during movement of irradiated fuel assemblies and core alterations because analysis indicates that isolation of the control room is not required for maintaining radiation exposure within acceptable limits following a fuel handling accident or gas decay tank rupture. Administrative controls address the role of the CRVS in maintaining control room habitability following an event at Indian Point Unit 2.
ACTIONS	<u>A.1</u> When one CRVS train is inoperable, action must be taken to restore OPERABLE status within 7 days. In this Condition, the remaining OPERABLE CRVS train is adequate to perform the control room protection function. However, the overall reliability is reduced because a failure in the OPERABLE CRVS train could result in loss of CRVS function. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and ability of the remaining train to provide the required capability. <u>B.1</u> When neither CRVS train is Operable, action must be taken to restore at least one train to OPERABLE status within 72 hours. The 72 hour Completion Time is acceptable because of the low probability of a DBA occurring during this time period.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

If Required Actions A.1 or B.1 are not met within the required Completion Time, the unit must be placed in a MODE that minimizes accident risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.11.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not too severe, testing each train once every month provides an adequate check of this system. Note that a CRVS train includes both the filter booster fan and an air-conditioning unit fan powered from the same safeguards power train. The 31 day Frequency is based on the reliability of the equipment and the two train redundancy availability.

SR 3.7.11.2

This SR verifies that the required CRVS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The CRVS filter tests are in accordance with the sections of Regulatory Guide 1.52 (Ref. 3) identified in the VFTP. The VFTP includes testing the performance of the HEPA filter, charcoal adsorber efficiency, minimum flow rate, and the physical properties of the activated charcoal. Specific test Frequencies and additional information are discussed in detail in the VFTP.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.11.3

This SR verifies that each CRVS train starts and operates on an actual or simulated actuation signal. The Frequency of 24 months is based on operating experience which has demonstrated this Frequency provides a high degree of assurance that the booster fans will operate and dampers actuate to the correct position when required.

SR 3.7.11.4

This SR verifies the integrity of the control room enclosure, and the assumed inleakage rates of the potentially contaminated air. The control room positive pressure, with respect to potentially contaminated adjacent areas, is periodically tested to verify proper functioning of the CRVS. During the operation in the incident mode with outside air makeup (i.e. 10% incident mode), the CRVS is designed to maintain the control room at a slight positive pressure with respect to adjacent areas in order to prevent unfiltered inleakage. The CRVS is designed to maintain this positive pressure with very low volumes of outside air makeup. Due to the difficulty of adjusting and maintaining the flow dampers to provide a low flow, it was determined that the damper should be adjusted to provide a flow of approximately 250 cfm (2.5% outside air makeup). Note that the higher the volume of outside air makeup to the Control Room, the higher the thyroid dose to the operators during an accident. The acceptance criteria of 400 cfm (4.0% outside air makeup) is the volume used in the Control Room dose assessment.

The SR Frequency of 24 months on a staggered test basis is acceptable because operating experience has demonstrated that the control room boundary is not normally disturbed. Staggered testing is acceptable because the SR is primarily a verification of Control Room integrity because fan operation is tested elsewhere.

(continued)

BASES (continued)

REFERENCES

1. FSAR, Section 9.9.
2. FSAR, Chapter 14.
3. Regulatory Guide 1.52, Rev. 2.
4. IP3 Technical Requirements Manual.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.7.11:
"Control Room Ventilation System (CRVS)"

PART 5:

NUREG-1431
Annotated to show differences between NUREG-1431 and ITS

Status of NUREG 1431 Generic Changes for ITS 3.7.11

This ITS Specification is based on NUREG-1431 Specification No. **3.7.10**
as modified by the following Generic Changes:

OG No.	TSTF No.	Generic Change Description	NRC STATUS	IP3 STATUS	JD No.
BWOG-009		ADD AN ACTION FOR LOSS OF CONTROL ROOM ENCLOSURE INTEGRITY	Rejected by TSTF	Not Incorporated	**N/A**
BWROG-017	**051 R0**	REVISE CONTAINMENT REQUIREMENTS DURING HANDLING IRRADIATED FUEL AND CORE ALTERATIONS (REQUIREMENTS LIMITED TO "RECENTLY" IRRADIATED FUEL)	NRC Review	Not Incorporated	**N/A**
WOG-086	**287 R2**	VENTILATION SYSTEM ENVELOPE ALLOWED OUTAGE TIME	NRC Review	Not Incorporated	**N/A**

B 3.7 PLANT SYSTEMS

B 3.7.10 Control Room Emergency Filtration System (CREFS)

BASES

BACKGROUND

The CREFS provides a protected environment from which operators can control the unit following an uncontrolled release of radioactivity[, chemicals, or toxic gas].

The CREFS consists of two independent, redundant trains that recirculate and filter the control room air. Each train consists of a prefilter or demister, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, and instrumentation also form part of the system, as well as demisters to remove water droplets from the air stream. A second bank of HEPA filters follows the adsorber section to collect carbon fines and provide backup in case of failure of the main HEPA filter bank.

The CREFS is an emergency system, parts of which may also operate during normal unit operations in the standby mode of operation. Upon receipt of the actuating signal(s), normal air supply to the control room is isolated, and the stream of ventilation air is recirculated through the system filter trains. The prefilters or demisters remove any large particles in the air, and any entrained water droplets present, to prevent excessive loading of the HEPA filters and charcoal adsorbers. Continuous operation of each train for at least 10 hours per month, with the heaters on, reduces moisture buildup on the HEPA filters and adsorbers. Both the demister and heater are important to the effectiveness of the charcoal adsorbers.

Actuation of the CREFS places the system in either of two separate states (emergency radiation state or toxic gas isolation state) of the emergency mode of operation, depending on the initiation signal. Actuation of the system to the emergency radiation state of the emergency mode of operation, closes the unfiltered outside air intake and unfiltered exhaust dampers, and aligns the system for recirculation of the control room air through the redundant trains of HEPA and the charcoal filters. The emergency radiation state also initiates pressurization and filtered ventilation of the air supply to the control room.

(continued)

NUREG-1431 Markup Inserts
ITS SECTION 3.7.11 - Control Room Ventilation System (CRVS)

INSERT: B 3.7-50-01

The Control Room Ventilation System consists of the following equipment: a single filter unit consisting of two roughing filters, two high efficiency particulate air (HEPA) filters; two activated charcoal adsorbers for removal of gaseous activity (principally iodines); two 100% capacity filter booster fans; and, a single duct system including dampers, controls and associated accessories to provide for three different air flow configurations. The air-conditioning units associated with the CRVS are governed by LCO 3.7.12, "Control Room Air Conditioning System (CRACS)."

The CRVS is divided into two trains with each train consisting of a filter booster fan and the associated inlet damper and the following components which are common to both trains: the control room filter unit, damper A (filter unit bypass for outside air makeup to the Control Room), damper B (filter unit inlet for outside air makeup to the Control Room), damper C (filter unit inlet for reticulated air), and the toilet and locker room exhaust fan. The two filter booster fans (F 31 and F 32) are powered from safeguards power trains 5A (EDG 33) and 6A (EDG 32), respectively. ~~Each of t~~The automatic dampers that are common to both trains ~~is~~ are positioned in the fail-safe position (open or closed) by either of the redundant actuation channels. R.1

NUREG-1431 Markup Inserts
ITS SECTION 3.7.11 - Control Room Ventilation System (CRVS)

INSERT: B 3.7-50-02

The three different CRVS air flow configurations are as follows:

a) Normal operation consists of approximately 85% (8500 cfm) unfiltered recirculated flow driven by the air-conditioning fans and approximately 15% (1500 cfm) unfiltered outside air makeup;

b) Incident mode with outside air makeup (i.e. 10% incident mode) consists of approximately 87% (9250 cfm) unfiltered recirculated flow driven by the two safety related air-conditioning fans, at least 10% (> 1000 cfm) filtered recirculated flow driven by either one of the two filter booster fans and approximately 2.5% to 4.0% (250 to 400 cfm) filtered outside air makeup;

c) Incident mode with no outside air makeup (i.e. 100% incident mode) consists of 85% (9100 cfm) unfiltered reticulated flow driven by the two safety related air-conditioning fans, approximately 15% filtered recirculated flow driven by either one of the two filter booster fans and no outside air makeup.

Note that the required recirculation rates are demonstrated with surveillance tests conducted with the air conditioning system (CRACS) operating. An inoperable CRACS fan will affect the flow balance of the CRVS due to interconnected ductwork. Therefore, if the fan associated with one of the air-conditioning units governed by LCO 3.7.12 is inoperable, Conditions in both LCO 3.7.11, Control Room Ventilation System, and LCO 3.7.12, Control Room Air Conditioning System (CRACS), will apply.

Incident mode with outside air makeup is the preferred method of operation during any radiological event because it provides outside air for pressurization of the Control Room. Calculations indicate that very low volumes of outside air makeup will maintain the Control Room at a slight positive pressure. Nevertheless, due to the difficulty of adjusting and maintaining the flow dampers to provide a low flow, it was determined that the damper should be adjusted to provide a flow of approximately 250 cfm (2.5% outside air makeup). However, a higher volume of outside air makeup to the Control Room increase the thyroid dose to the operators during an accident. Therefore, the Control Room dose assessment assumes a filtered outside air makeup of approximately 400 cfm (4.0% outside air makeup).

INSERT: B 3.7-50-02 (continued)

On a Safety Injection signal or high radiation in the Control Room (Radiation Monitor R-1), the CRVS will actuate to the incident mode with outside air makeup (i.e. 10% incident mode). This will cause one of the two filters booster fans to start, the locker room exhaust fan to stop, and CRVS dampers to open or close as necessary to filter all incoming outside air and direct approximately 10% of the recirculated air through the filter unit. In the event that the first booster fan fails to start, the second booster fan will start after a predetermined time delay.

If for any reason it is required or desired to operate with 100% recirculated air (e.g., toxic gas condition is identified), the CRVS can be placed in the incident mode with no outside air makeup (i.e. 100% incident mode) by remote manually operated switches. The Firestat detectors will also initiate 100% incident mode in the CRVS.

BASES

BACKGROUND (continued)

Outside air is filtered, diluted with building air from the electrical equipment and cable spreading rooms, and added to the air being recirculated from the control room. Pressurization of the control room prevents infiltration of unfiltered air from the surrounding areas of the building. The actions taken in the toxic gas isolation state are the same, except that the signal switches control room ventilation to an isolation alignment to prevent outside air from entering the control room.

Insert: B3.7-51-01

The air entering the control room is continuously monitored by radiation and toxic gas detectors. One detector output above the setpoint will cause actuation of the emergency radiation state or toxic gas isolation state, as required. H DB.1

Insert B 3.7-51-02

Insert: B3.7-51-03

The actions of the toxic gas isolation state are more restrictive, and will override the actions of the emergency radiation state.

Insert: B3.7-51-04

A single train will pressurize the control room to about [0.125] inches water gauge. The CREFS CRVS operation in maintaining the control room habitable is discussed in the FSAR, Section [6.4] 9.9 (Ref. 1).

Redundant supply and recirculation trains provide the required filtration should an excessive pressure drop develop across the other filter train. Normally open isolation dampers are arranged in series pairs so that the failure of one damper to shut will not result in a breach of isolation. The CREFS CRVS is designed in accordance with Seismic Category I requirements.

The CREFS is designed to maintain the control room environment for 30 days of continuous occupancy after a Design Basis Accident (DBA) without exceeding a 5 rem whole body dose or its equivalent to any part of the body 30 rem to the thyroid.

APPLICABLE SAFETY ANALYSES

The CREFS CRVS components are arranged building in redundant, safety related active ventilation trains. The location of components and ducting within the control room envelope ensures an adequate supply of filtered air to all areas requiring access. The CREFS CRVS provides airborne radiological protection for the control room operators, as demonstrated by the control room accident dose analyses for the most limiting design basis R.1

(continued)

NUREG-1431 Markup Inserts
ITS SECTION 3.7.11 - Control Room Ventilation System (CRVS)

INSERT: B 3.7-51-01

On a Safety Injection signal or high radiation in the Control Room (Radiation Monitor R-1),

INSERT: B 3.7-51-02

of the CRVS (i.e., incident mode with outside air makeup (i.e. 10% incident mode)).

INSERT: B 3.7-51-03

The CRVS does not actuate automatically in response to toxic gases. Separate chlorine, ammonia and oxygen probes are provided to detect the presence of these gases in the outside air intake. Additionally, monitors in the Control Room will detect low oxygen levels and high levels of chlorine and ammonia. The CRVS may be placed in the incident mode with no outside air makeup (i.e. 100% incident mode) to respond to these conditions. Instrumentation for toxic gas monitoring is governed by the IP3 Technical Requirements Manual (TRM) (Ref. 4). Generally, the manually initiated

INSERT: B 3.7-51-04

create a slight positive pressure in the control room.

BASES

APPLICABLE SAFETY ANALYSES (continued)

loss of coolant accident, fission product release presented in the FSAR, Chapter 15 (Ref. 2). 14

Insert: B 3.7-52-01

Insert: B 3.7-52-02

The analysis of toxic gas releases demonstrates that the toxicity limits are not exceeded in the control room following a toxic chemical release, as presented in Reference 1. DB1 DB2

Insert: B 3.7-52-03

The worst case single active failure of a component of the CREFS assuming a loss of offsite power, does not impair the ability of the system to perform its design function.

CRVS

The CREFS satisfies Criterion 3 of the NRC Policy Statement. 10 CFR 50.36

LCO

CRVS

Two independent and redundant CREFS trains are required to be OPERABLE to ensure that at least one is available. assuming a single failure disables the other train. Total system failure could result in exceeding a dose of 5 rem to the control room operator in the event of a large radioactive release.

Insert: B3.7-52-04

The CREFS is considered OPERABLE when the individual components necessary to limit operator exposure are OPERABLE in both trains. A CREFS train is OPERABLE when the associated:

CRVS

Insert: B 3.7-52-05

a. Fan is OPERABLE;

b. HEPA filters and charcoal adsorbers are not excessively restricting flow, and are capable of performing their filtration functions; and

c. Heater, demister, ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained. DB1

or in the incident mode

In addition, the control room boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors.

Insert: B 3.7-52-06

APPLICABILITY

In MODES 1, 2, 3, 4, [5, and 6,] and during movement of irradiated fuel assemblies [and during CORE ALTERATIONS].

(continued)

INSERT: B 3.7-52-01

accident (i.e., DBA LOCA)

INSERT: B 3.7-52-02

Radiation monitor R-1 is not required for the Operability of the Control Room Ventilation System because control room isolation is initiated by the safety injection signal in MODES 1, 2, 3, and 4 and control room isolation is not required for maintaining radiation exposure within General Design Criteria 19 limits following a fuel handling accident or gas-decay-tank rupture.

INSERT: B 3.7-52-03

However, the original CRVS design was not required to meet single failure criteria and, although upgraded from the original design, CRVS does not satisfy all requirements in IEEE-279 for single failure tolerance. Each of the automatic dampers that are common to both trains is positioned in the fail-safe position (open or closed) by either of the redundant actuation channels.

INSERT: B 3.7-52-04

5 rem whole body or 30 rem to the thyroid of

INSERT: B 3.7-52-05

F
~~a~~ ~~f~~ilter booster fan and an air-conditioning unit fan powered from the same safeguards power train are

R.1

NUREG-1431 Markup Inserts
ITS SECTION 3.7.11 - Control Room Ventilation System (CRVS)

INSERT: B 3.7-52-06

Instrumentation for toxic gas monitoring is governed by the IP3 Technical Requirements Manual (TRM) (Ref. 4) and is not included in the LCO.

Note that the required recirculation rates are demonstrated with surveillance tests conducted with the air conditioning system (CRACS) operating. An inoperable CRACS fan will affect the flow balance of the CRVS due to interconnected ductwork. Therefore, if the fan associated with one of the air-conditioning units governed by LCO 3.7.12 is inoperable, Conditions in both LCO 3.7.11, Control Room Ventilation System, and LCO 3.7.12, Control Room Air Conditioning System (CRACS), will apply.

BASES

APPLICABILITY (continued)

(CRVS) ~~CREFS~~ must be OPERABLE to (limit) ~~control~~ operator exposure during and following a DBA.

Insert: B3.7-53-01

~~In [MODE 5 or 6], the CREFS is required to cope with the release from the rupture of an outside waste gas tank.~~

~~During movement of irradiated fuel assemblies [and CORE ALTERATIONS], the CREFS must be OPERABLE to cope with the release from a fuel handling accident.~~

ACTIONS

A.1

When one (CRVS) ~~CREFS~~ train is inoperable, action must be taken to restore OPERABLE status within 7 days. In this Condition, the remaining OPERABLE ~~CREFS~~ train is adequate to perform the control room protection function. However, the overall reliability is reduced because a ~~single~~ failure in the OPERABLE ~~CREFS~~ train could result in loss of ~~CREFS~~ function. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and ability of the remaining train to provide the required capability.

Insert: B3.7-53-02

(C) ~~B~~.1 and ~~B~~.2

Insert: B3.7-53-03

~~In MODE 1, 2, 3, or 4, if the inoperable CREFS train cannot be restored to OPERABLE status~~ within the required Completion Time, the unit must be placed in a MODE that minimizes accident risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

~~C.1, C.2.1, and C.2.2~~

~~[In MODE 5 or 6, or] during movement of irradiated fuel assemblies [, or during CORE ALTERATIONS], if the inoperable CREFS train cannot be restored to OPERABLE status within the required Completion Time, action must be taken to immediately place the OPERABLE CREFS train in the emergency~~

~~(continued)~~

NUREG-1431 Markup Inserts
ITS SECTION 3.7.11 - Control Room Ventilation System (CRVS)

INSERT: B 3.7-53-01

The CRVS is not required in MODE 5 or 6, or during movement of irradiated fuel assemblies and core alterations because analysis indicates that isolation of the control room is not required for maintaining radiation exposure within acceptable limits following a fuel handling accident or gas decay tank rupture.

Administrative controls address the role of the CRVS in maintaining control room habitability following an event at Indian Point Unit 2.

INSERT: B 3.7-53-02

B.1

When neither CRVS train is Operable, action must be taken to restore at least one train to OPERABLE status within 72 hours. The 72 hour Completion Time is acceptable because of the low probability of a DBA occurring during this time period.

INSERT: B 3.7-53-03

If Required Actions A.1 or B.1 are not met

BASES

ACTIONS

C.1, C.2.1, and C.2.2 (continued)

mode. This action ensures that the remaining train is OPERABLE, that no failures preventing automatic actuation will occur, and that any active failure would be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that could result in a release of radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk. This does not preclude the movement of fuel to a safe position.

Required Action C.1 is modified by a Note indicating to place the system in the toxic gas protection mode if automatic transfer to toxic gas protection mode is inoperable.

D.1 and D.2

[In MODE 5 or 6, or] during movement of irradiated fuel assemblies [, or during CORE ALTERATIONS], with two CREFS trains inoperable, action must be taken immediately to suspend activities that could result in a release of radioactivity that might enter the control room. This places the unit in a condition that minimizes accident risk. This does not preclude the movement of fuel to a safe position.

E.1

If both CREFS trains are inoperable in MODE 1, 2, 3, or 4, the CREFS may not be capable of performing the intended function and the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1 (11)

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not too severe,

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1 (continued)

testing each train once every month provides an adequate check of this system. Monthly heater operations dry out any moisture accumulated in the charcoal from humidity in the ambient air. [Systems with heaters must be operated for ≥ 10 continuous hours with the heaters energized. Systems without heaters need only be operated for ≥ 15 minutes to demonstrate the function of the system.] The 31 day Frequency is based on the reliability of the equipment and the two train redundancy availability.

SR 3.7.10.2

This SR verifies that the required CREFS testing is performed in accordance with the [Ventilation Filter Testing Program (VFTP)]. The CREFS filter tests are in accordance with Regulatory Guide 1.52 (Ref. 3). The [VFTP] includes testing the performance of the HEPA filter, charcoal adsorber efficiency, minimum flow rate, and the physical properties of the activated charcoal. Specific test Frequencies and additional information are discussed in detail in the [VFTP].

SR 3.7.10.3

This SR verifies that each CREFS train starts and operates on an actual or simulated actuation signal. The Frequency of [18] months is specified in Regulatory Guide 1.52 (Ref. 3).

SR 3.7.10.4

This SR verifies the integrity of the control room enclosure, and the assumed inleakage rates of the potentially contaminated air. The control room positive pressure, with respect to potentially contaminated adjacent areas, is periodically tested to verify proper functioning of the CREFS. During the emergency mode of operation, the CREFS is designed to pressurize the control room ≥ [0.125] inches water gauge positive pressure with respect to adjacent areas in order to prevent unfiltered inleakage. The CREFS is designed to maintain this positive pressure

(continued)

INSERT: B 3.7-55-01

based on operating experience which has demonstrated this Frequency provides a high degree of assurance that the booster fans will operate and dampers actuate to the correct position when required.

INSERT: B 3.7-55-02

operation in the incident mode with outside air makeup (i.e. 10% incident mode)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.10.4 (continued)

11

Insert: B3.7-56-01

with one train at a makeup flow rate of [3000] cfm. The Frequency of [18] months on a STAGGERED TEST BASIS is consistent with the guidance provided in NUREG-0800 (Ref. 4).

REFERENCES

1. FSAR, Section (6.4). 9.9
2. FSAR, Chapter (15). 14
3. Regulatory Guide 1.52, Rev. 2.
4. ~~NUREG-0800, Section 6.4, Rev. 2, July 1981.~~

Insert: B3.7-56-02

NUREG-1431 Markup Inserts
ITS SECTION 3.7.11 - Control Room Ventilation System (CRVS)

INSERT: B 3.7-56-01

with very low volumes of outside air makeup. Due to the difficulty of adjusting and maintaining the flow dampers to provide a low flow, it was determined that the damper should be adjusted to provide a flow of approximately 250 cfm (2.5% outside air makeup). Note that the higher the volume of outside air makeup to the Control Room, the higher the thyroid dose to the operators during an accident. The acceptance criteria of 400 cfm (4.0% outside air makeup) is the volume used in the Control Room dose assessment.

The SR Frequency of 24 months on a staggered test basis is acceptable because operating experience has demonstrated that the control room boundary is not normally disturbed. Staggered testing is acceptable because the SR is primarily a verification of Control Room integrity because fan operation is tested elsewhere.

INSERT: B 3.7-56-02

4. IP3 Technical Requirements Manual

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.7.12:
"Control Room Air Conditioning System (CRACS)"

PART 1:

Indian Point 3
Improved Technical Specifications and Bases

Rev 1 affects Bases page only

BASES

ACTIONS (continued)

B.1

When neither CRACS train is Operable, action must be taken to restore at least one train to OPERABLE status within 72 hours. The 72 hour Completion Time is acceptable because of the low probability of a DBA occurring during this time period and because alternate nonsafety cooling means are typically available.

C.1 and C.2

If Required Actions A.1 or B.1 are not met within the required Completion Time, the unit must be placed in a MODE that minimizes the risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1

This SR verifies that the heat removal capability of the system is sufficient to remove the heat load required to maintain functional capacity of the Control Room at all times (Ref. 1). This SR consists of a combination of testing and calculations. The 24 month Frequency is appropriate since significant degradation of the CRACS is slow and is not expected over this time period.

RAI-03

REFERENCES

1. FSAR, Section 9.9.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.7.12:
"Control Room Air Conditioning System (CRACS)"

PART 5:

NUREG-1431
Annotated to show differences between
NUREG-1431 and ITS

Status of NUREG 1431 Generic Changes for ITS 3.7.12

This ITS Specification is based on NUREG-1431 Specification No. **3.7.11** as modified by the following Generic Changes:

OG No.	TSTF No.	Generic Change Description	NRC STATUS	IP3 STATUS	JD No.
BWROG-017	**051 R0**	REVISE CONTAINMENT REQUIREMENTS DURING HANDLING IRRADIATED FUEL AND CORE ALTERATIONS (REQUIREMENTS LIMITED TO "RECENTLY" IRRADIATED FUEL)	NRC Review	Not Incorporated	**N/A**

BASES

ACTIONS (continued)

E.1

If both CREATCS trains are inoperable in MODE 1, 2, 3, or 4, the control room CREATCS may not be capable of performing its intended function. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.7.11.1 12

Insert B 3.7-60-01

This SR verifies that the heat removal capability of the system is sufficient to remove the heat load assumed in the [safety analyses] in the control room. This SR consists of a combination of testing and calculations. The [18] 24 month Frequency is appropriate since significant degradation of the CREATCS CRACS is slow and is not expected over this time period.

REFERENCES

1. FSAR, Section [6.4]. 9.9

NUREG-1431 Markup Inserts
ITS SECTION 3.7.12 - Control Room Air Conditioning System (CRACS)

INSERT: B 3.7-60-01

required to maintain functional capacity of the Control Room at all times (Ref. 1).

R.1

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.7.15: "Spent Fuel Pit Boron Concentration"

PART 1:

Indian Point 3
Improved Technical Specifications and Bases

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.15.1	Verify the spent fuel pit boron concentration is within limit.	31 days

NYPA

BASES

APPLICABLE SAFETY ANALYSES (continued)

The concentration of dissolved boron in the spent fuel pit satisfies Criterion 2 of 10 CFR 50.36.

LCO

The spent fuel pit boron concentration is required to be ≥ 1000 ppm. The specified concentration of dissolved boron in the spent fuel pit preserves the assumptions used in the analyses of the potential critical accident scenarios as described in Reference 3. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the spent fuel pit until a spent fuel pit verification confirms that there are no mis-loaded fuel assemblies. With no mis-loaded fuel assemblies and unborated water, the spent fuel pit design is sufficient to maintain the core at $k_{eff} \leq 0.95$.

APPLICABILITY

This LCO applies whenever fuel assemblies are stored in the spent fuel pit, until a complete spent fuel pit verification has been performed following the last movement of fuel assemblies in the spent fuel pit. This LCO does not apply following the verification, since the verification would confirm that there are no misloaded fuel assemblies. With no further fuel assembly movements in progress, there is no potential for a misloaded fuel assembly or a dropped fuel assembly.

NYPA

ACTIONS

A.1, A.2.1 and A.2.2

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply.

When the concentration of boron in the spent fuel pit is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. The concentration of boron is restored simultaneously with suspending

(continued)

BASES

ACTIONS

A.1, A.2.1 and A.2.2

movement of fuel assemblies. Alternatively, beginning a verification of the Spent Fuel Pit fuel locations, to ensure proper locations of the fuel, can be performed. However, prior to resuming movement of fuel assemblies, the concentration of boron must be restored. This does not preclude movement of a fuel assembly to a safe position.

If the LCO is not met while moving irradiated fuel assemblies in MODE 5 or 6, LCO 3.0.3 would not be applicable. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.15.1

This SR verifies that the concentration of boron in the spent fuel pit is within the required limit. As long as this SR is met, the analyzed accidents are fully addressed. The 31 day Frequency is appropriate because no major replenishment of spent fuel pit water is expected to take place over such a short period of time. This SR is not required to be met or performed if a spent fuel pit verification for conformance with LCO 3.7.16, Figures 3.7.16-1 and B 3.7.16-1, has been performed on all fuel assemblies since the last verification following the last movement of fuel assemblies in the spent fuel pit.

NYPA

RAI-1

REFERENCES

1. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).

(continued)

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.7.15:
"SPENT FUEL PIT BORON CONCENTRATION"

PART 2:

CURRENT TECHNICAL SPECIFICATION PAGES

Annotated to show differences between CTS and ITS

CTS PAGE	AMENDMENT FOR REV O SUBMITTAL	AMENDMENT FOR REV 1 SUBMITTAL	COMMENT
3.8-3	114	114	
3.8-5	173	173	
3.8-6	175	189	Deleted Requirement of 267 Hours for Discharge of > 76 Assemblies. No impact on 3.7.15.
3.8-7	173	189	Deleted FSAR Reference in Bases Page. No impact on 3.7.15.
T4.1-2(1)	139	200	Deleted Boric Acid Tank Sampling Requirement Frequencies No impact on 3.7.15
5.4-1	173	173	

TABLE 4.1-2 (Sheet 1 of 2)

FREQUENCIES FOR SAMPLING TESTS

Sample	Analysis	Frequency	Maximum Time Between Analysis
1. Reactor Coolant	Gross Activity[1]	5 days/week[1][4]	3 days[4]
	Tritium Activity	Weekly[1]	10 days
	Boron concentration	2 days/week	5 days
	Radiochemical (gamma)[2] Spectral Check	Monthly	45 days
	Oxygen and Chlorides Concentration	3 times per 7 days	3 days
	Fluorides Concentration	Weekly	10 days
	Ē Determination [3]	Semi-Annually	30 weeks
	Isotopic Analysis for I-131, I-133, I-135	Once per 14 days[5]	20 days
2. Boric Acid Tank	Boron Concentration, Chlorides	Weekly	10 days
3. Spray Additive Tank	NaOH Concentration	Monthly	45 days
4. Accumulators	Boron Concentration	Monthly	45 days
5. Refueling Water Storage Tank	Boron Concentration pH, Chlorides	Monthly	45 days
	Gross Activity	Quarterly	16 weeks
6. Secondary Coolant	I-131 Equivalent (Isotopic Analysis)	Monthly	45 days
	Gross Activity	3 times per 7 days	3 days
7. Component Cooling Water	Gross Activity, Corrosion Inhibitor and pH	Monthly	45 days
8. Spent Fuel Pool (when fuel stored)	Gross Activity Boron Concentration, Chlorides	Monthly	45 days

Amendment No 139

SEE CTS MASTER MARKUP

SR 3.7.15.1

SEE RELOCATED CTS

31 days unless pool verification completed

L.1

A.3

CLB.1

See Amendment 200, next page

No impact on ITS 3.7.15

ITS 3.7.15

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.7.15: "Spent Fuel Pit Boron Concentration"

PART 3:

DISCUSSION OF CHANGES

Differences between CTS and ITS

DISCUSSION OF CHANGES
ITS SECTION 3.7.15 - SPENT FUEL PIT BORON CONCENTRATION

MORE RESTRICTIVE

None

LESS RESTRICTIVE

L.1 CTS 3.8.C.3 specifies the Applicability of requirements for spent fuel pit boron concentration as during periods of spent fuel movement in the spent fuel pit. CTS 5.4.3 specifies the Applicability of requirements for spent fuel pit boron concentration as whenever there is fuel in the pit (See ITS 3.7.15, DOC A.3). CTS 4.1-2, Item 8, requires monthly verification of boron concentration "when fuel stored" in the spent fuel pit. Therefore, IP3 requirements for spent fuel pit boron concentration apply whenever there is fuel in the spent fuel pit.

ITS 3.7.15 specifies that spent fuel pit boron concentration limits must be met "When fuel assemblies are stored in the fuel storage pit and a spent fuel pit verification has not been performed since the last movement of fuel assemblies in the fuel storage pit." This change is acceptable because, as stated in the SER Related to IP3 CTS Amendment 173, April 15, 1997, the spent fuel pit design is sufficient to maintain the core at $k_{eff} \leq 0.95$ with unborated water if there are no misloaded fuel assemblies (i.e., fuel storage locations meet the requirements of ITS LCO 3.7.16). The requirement in ITS SR 3.7.16.1 to verify proper storage location of each fuel assembly prior to landing the assembly in the spent fuel pit and the requirement to perform a complete spent fuel pit verification prior to relaxing boron concentration requirements provides a high degree of assurance that boron concentration will be maintained at all times there is a potential for a mis-positioned fuel assembly. Finally, even if requirements for boron concentration are eliminated, spent fuel pit boron concentration is not reduced after the completion of fuel movement and is usually maintained > 1900 ppm. As a result, the practical consequence of this change is to eliminate the formal requirement in CTS Table 4.1-2, Item 8, for monthly verification of spent fuel pit boron concentration after a storage location verification is completed. Therefore, eliminating the requirements for a minimum boron concentration in the spent fuel pit if no fuel movement is in progress and a storage verification was performed after the last movement of fuel assemblies will not significantly increase the potential for criticality in the spent fuel pit. This change has no significant adverse impact on safety.

REMOVED DETAIL

None

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.7.15:
"Spent Fuel Pit Boron Concentration"

PART 4:

No Significant Hazards Considerations
for
Changes between CTS and ITS
that are
Less Restrictive

No Significant Hazard Considerations for Changes that are Administrative, More Restrictive, and Removed Details are the same for all Packages. A Copy is included at the end of the Package.

NO SIGNIFICANT HAZARDS EVALUATION
ITS SECTION 3.7.15 - Spent Fuel Pit BORON CONCENTRATION

LESS RESTRICTIVE
("L.1" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

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

   This change eliminates the requirement for maintaining a minimum boron concentration in the spent fuel pit if no fuel movement is in progress and a storage verification was performed after the last movement of fuel assemblies. This change will not result in a significant increase in the probability or consequences of an accident previously evaluated because , as stated in the SER Related to IP3 CTS Amendment 173, April 15, 1997, the spent fuel pit design is sufficient to maintain the core at $k_{eff} \leq 0.95$ with unborated water if there are no misloaded fuel assemblies (i.e., fuel storage locations meet the requirements of ITS LCO 3.7.16). The requirement in ITS SR 3.7.16.1 to verify proper storage location of each fuel assembly prior to landing the assembly in the spent fuel pit and the requirement to perform a complete spent fuel pit verification prior to relaxing boron concentration requirements provides a high degree of assurance that boron concentration will be maintained at all times there is a potential for a mis-positioned fuel assembly. Finally, even if requirements for boron concentration are eliminated, spent fuel pit boron concentration is not reduced after the completion of fuel movement and is usually maintained > 1900 ppm. As a result, the practical consequence of this change is to eliminate the formal requirement in CTS Table 4.1-2, Item 8, for monthly verification of spent fuel pit boron concentration after a storage location verification is completed. Therefore, eliminating the requirements for a minimum boron concentration in the spent fuel pit if no fuel movement is in progress and a storage verification was performed after the last movement of fuel assemblies will not significantly increase the potential for criticality in the spent fuel pit.

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

   The proposed change will not involve any physical changes to systems, structures, or components, or involve a change in normal plant operation. Therefore, it will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   This change does not involve a significant reduction in a margin of safety because, as stated in the SER Related to IP3 CTS Amendment 173, April 15, 1997, the spent fuel pit design is sufficient to maintain the core at $k_{eff} \leq 0.95$ with unborated water if there are no misloaded fuel assemblies (i.e., fuel storage locations meet the requirements of ITS LCO 3.7.16). The requirement in ITS SR 3.7.16.1 to verify proper storage location of each fuel assembly prior to landing the assembly in the spent fuel pit and the requirement to perform a complete spent fuel pit verification prior to relaxing boron concentration requirements provides a high degree of assurance that boron concentration will be maintained at all times there is a potential for a mis-positioned fuel assembly. Finally, even if requirements for boron concentration are eliminated, spent fuel pit boron concentration is not reduced after the completion of fuel movement and is usually maintained at 1900 ppm. As a result, the practical consequence of this change is to eliminate the formal requirement in CTS Table 4.1-2, Item 8, for monthly verification of spent fuel pit boron concentration after a storage location verification is completed. Therefore, eliminating the requirements for a minimum boron concentration in the spent fuel pit if no fuel movement is in progress and a storage verification was performed after the last movement of fuel assemblies will not significantly increase the potential for criticality in the spent fuel pit.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.7.15:
"Spent Fuel Pit Boron Concentration"

PART 5:

NUREG-1431
Annotated to show differences between NUREG-1431 and ITS

Status of NUREG 1431 Generic Changes for ITS 3.7.15

This ITS Specification is based on NUREG-1431 Specification No. **3.7.16**
as modified by the following Generic Changes:

OG No.	TSTF No.	Generic Change Description	NRC STATUS	IP3 STATUS	JD No.
CEOG-023 R1	**070 R1**	FUEL STORAGE POOL VERIFICATION	APPROVED/INCORPORATED	Incorporated	**T.1**
CEOG-051	**139 R1**	INCORRECT CRITERIA DEFINED IN B 3.7.16	APPROVED/INCORPORATED	Incorporated	**T.2**

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.~~16~~ 15.1 Verify the ~~fuel storage pool~~ spent fuel pit boron concentration is within limit.	~~7~~ 31 days

⟨Table 4.1-2, #8⟩

~~⟨DOC M.1⟩~~
⟨DOC L.1⟩

R.1

R.1

CLB.1 R.1

BASES

APPLICABILITY (continued)

progress, there is no potential for a misloaded fuel assembly or a dropped fuel assembly.

ACTIONS

A.1, A.2.1, and A.2.2

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply.

When the concentration of boron in the fuel storage pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. The concentration of boron is restored simultaneously with suspending movement of fuel assemblies. ~~An acceptable alternative is to verify by administrative means that the fuel storage pool verification has been performed since the last movement of fuel assemblies in the fuel storage pool.~~ However, prior to resuming movement of fuel assemblies, the concentration of boron must be restored. This does not preclude movement of a fuel assembly to a safe position.

Insert: B3.7-83-01

T.1

If the LCO is not met while moving irradiated fuel assemblies in MODE 5 or 6, LCO 3.0.3 would not be applicable. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.16.1 (15)

This SR verifies that the concentration of boron in the fuel storage pool is within the required limit. As long as this SR is met, the analyzed accidents are fully addressed. The 31 (7) day Frequency is appropriate because no major replenishment of pool water is expected to take place over such a short period of time.

1 R.1 <CLB.1>

Insert: B3.7-83-02

(continued)

NUREG-1431 Markup Inserts
ITS SECTION 3.7.15 - SPENT FUEL PIT BORON CONCENTRATION

INSERT: B 3.7-83-01

Alternatively, beginning a verification of the Spent Fuel Pit fuel locations, to ensure proper locations of the fuel, can be performed.

INSERT: B 3.7-83-02

This SR is not required to be met or performed if a spent fuel pit verification for conformance with LCO 3.7.16, Figures 3.7.16-1 and B 3.7.16-1, has been performed on all fuel assemblies since the last verification following the last movement of fuel assemblies in the spent fuel pit.

R.1

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.7.15:
"Spent Fuel Pit Boron Concentration"

PART 6:

Justification of Differences between

NUREG-1431 and IP3 ITS

JUSTIFICATION OF DIFFERENCES FROM NUREG-1431
ITS SECTION 3.7.15 - SPENT FUEL PIT BORON CONCENTRATION

RETENTION OF EXISTING REQUIREMENT (CURRENT LICENSING BASIS)

CLB.1 CTS Table 4.1-2, Item 8 requires a monthly verification of spent fuel pit boron concentration whenever fuel is stored in the spent fuel pit. NUREG-1431, STS SR 3.7.16 requires verification of spent fuel pit boron concentration every 7 days unless a complete fuel location verification is completed. ITS SR 3.7.15 will maintain the current license basis of performing this surveillance every 31 days. NYPA would normally option to maintain the minimum boron concentration all the time because it is an onerous task to perform the complete verification of all the spent fuel assemblies due to the numerous spent fuel assemblies. Maintaining the current license basis of performing this surveillance every 31 days is needed because a 7-day Frequency is a burden without an appreciable improvement to safety. Normally the spent fuel pit is maintained at ≥ 1900 PPM of boron concentration that is well above the minimum required before movement of irradiated fuel assemblies is allowed by ITS 3.7.15 (i.e., ≥ 1000 PPM). There is a high degree of assurance that the spent fuel pit minimum boron cocnentration will be maintained above the LCO limits because it is very unlikely that a significant dilution of the borated water in the spent fuel pit would take place over this period of time. This change has no significant adverse impact on safety.

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT

PA.1 Corrected typographical error or made a minor editorial improvement to improve clarity and ensure requirements are fully understood and consistently applied. There are no technical changes to requirements as specified in NUREG 1431, Rev. 1; therefore, this change is not a significant or generic deviation from NUREG 1431, Rev 1.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN OR DESIGN BASIS

DB.1 Design or implementation details are incorporated or revised as necessary to more precisely describe IP3 current design or practice. These changes are intended to describe the design, improve clarity, or ensure requirements are fully understood and consistently applied. Unless identified and described below, these changes are self-

Indian Point 3
Improved Technical Specifications (ITS) Conversion Package

Technical Specification 3.7.16:
"Spent Fuel Assembly Storage"

PART 5:

NUREG-1431
Annotated to show differences between NUREG-1431 and ITS

Status of NUREG 1431 Generic Changes for ITS 3.7.16

This ITS Specification is based on NUREG-1431 Specification No. **3.7.17**
as modified by the following Generic Changes:

OG No.	TSTF No.	Generic Change Description	NRC STATUS	IP3 STATUS	JD No.
N/A	N/A	NO GENERIC CHANGES ARE POSTED AGAINST THIS SPECIFICATION.	Not Applicable	Not Applicable	N/A

NUREG-1431 Markup Inserts
ITS SECTION 3.7.16 - SPENT FUEL ASSEMBLY STORAGE

INSERT: B 3.7-85-02

and fuel storage locations, enrichment and burnup are in conformance with analysis assumptions and this LCO.

INSERT: B 3.7-85-03

concentration is verified to be within the limits specified in LCO 3.7.15, Spent Fuel Pit Boron Concentration, prior to movement of ~~fuel assemblies in the spent fuel pit~~ any fuel assembly.

1 R.1

INSERT: B 3.7-85-04

(i.e., verification that the spent fuel pit boron concentration is within limit).

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.7.17:
"Secondary Specific Activity"

PART 3:

DISCUSSION OF CHANGES

Differences between CTS and ITS

limit in Modes 1, 2, 3 and 4. In conjunction with the change in applicability, ITS 3.7.17, Required Action A.1 and A.2 require that the unit be placed in and Mode 5 (i.e., outside the expanded Applicability) if the specific activity is not within the limit.

This change, adding Mode 4 to the Applicability for secondary system specific activity, is needed because SG venting to the atmosphere may be needed to remove decay heat in Mode 4. This change is acceptable because it does not introduce any operation which is un-analyzed while requiring that the LCO for secondary system specific activity is applicable whenever there is a potential that SG venting to the atmosphere may be needed to remove decay heat. Therefore, this change has no adverse impact on safety.

M.2 CTS Table 4.1-2, Item 6, requires verification that secondary system specific activity (i.e., Dose Equivalent I-131) is within specified limits every month with a maximum time between secondary coolant specific activity analyses of 45 days. ITS 3.7.17 maintains the requirement to verify secondary system specific activity every 31 days but limits the maximum time between analyses based on ITS SR 3.0.2 which allows a 25% grace period for a maximum interval of approximately 7.5 days. This change is not needed to satisfy technical requirements but is being adopted for consistency with the NUREG-1431 and to simplify application of ITS SR 3.0.2. This change has no impact on safety.

LESS RESTRICTIVE

L.1 CTS Table 4.1-2, Item 6 (second part), requires checking secondary system gross activity at least 3 times every 7 days with a maximum time between secondary gross activity analyses of 3 days. ITS 3.7.17, Secondary Specific Activity, does not have any requirements for periodic verification of secondary side gross activity. Deletion of the requirement for periodic verification of secondary side gross activity is acceptable because limitations on SG gross activity is not an assumption of any accident analysis and the information provided by this test (i.e., early identification of SG tube leakage and significant changes in RCS activity) is provided to the operators by other methods (e.g., air ejector monitors). Therefore, deletion of this requirement has no adverse impact on safety.

REMOVED DETAIL

None

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.7.17:
"Secondary Specific Activity"

PART 4:

No Significant Hazards Considerations
for
Changes between CTS and ITS
that are
Less Restrictive

No Significant Hazard Considerations for Changes that are Administrative, More Restrictive, and Removed Details are the same for all Packages. A Copy is included at the end of the Package.

NO SIGNIFICANT HAZARDS EVALUATION
ITS SECTION 3.7.17 - SECONDARY SPECIFIC ACTIVITY

LESS RESTRICTIVE
<u>("L.1" Labeled Comments/Discussions)</u>

York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

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

   CTS Table 4.1-2, Item 6 (second part), requires checking secondary system gross activity at least 3 times every 7 days with a maximum time between secondary gross activity analyses of 3 days. ITS 3.7.17, Secondary Specific Activity, does not have any requirements for periodic verification of secondary side gross activity.
   This change will not result in a significant increase in the probability or consequences of an accident previously evaluated because limitations on SG gross activity is not an assumption of any accident analysis and the information provided by this test (i.e., early identification of SG tube leakage and significant changes in secondary side activity) is provided to the operators by other methods (e.g., air ejector monitors).

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

   The proposed change will not involve any physical changes to systems, structures, or components, or involve a change in normal plant operation. Therefore, it will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   This change does not involve a significant reduction in a margin of safety because secondary side gross activity is not an assumption in any accident analysis and the information provided by sampling (i.e., early identification of SG tube leakage and significant changes in secondary side activity) is provided to the operators by other methods (e.g., air ejector monitors).

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.9.2: "Nuclear Instrumentation"

PART 3:

DISCUSSION OF CHANGES

Differences between CTS and ITS

DISCUSSION OF CHANGES
ITS SECTION 3.9.2 - NUCLEAR INSTRUMENTATION

ADMINISTRATIVE

A.1 In the conversion of the Indian Point Unit 3 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Additionally, editorial changes, reformatting, and revised numbering are adopted to make ITS consistent with the conventions in NUREG-1431, Standard Technical Specifications, Westinghouse Plants, Rev. 1, i.e., the improved Standard Technical Specifications.

The CTS Bases are deleted and replaced with comprehensive ITS Bases designed to support interpretation and implementation of the associated Technical Specifications. The Bases explain, clarify, and document the reasons (i.e., bases) for the associated Technical Specifications, and reflect the IP3 plant specific design, analyses, and licensing basis. In accordance with 10 CFR 50.36(a), the ITS Bases are included with the proposed ITS conversion application; however, deletion of the CTS Bases and the adoption of the ITS Bases is an administrative change with no impact on safety.

A.2 CTS Limiting Conditions for Operation (LCOs) and Surveillance Requirements (SRs) include statements of the objective and the applicability. The CTS statements of objective and applicability are deleted because these statements do not establish any requirements and do not provide any guidance for the application of CTS requirements. Therefore, deletion of these statements has no significant adverse impact on safety.

A.3 CTS 3.8.A.4 requires that the core subcritical neutron flux be continuously monitored by two source range neutron monitors whenever core geometry is being changed. When core geometry is not being changed, at least one source range neutron flux monitor shall be in service. CTS 3.8.B requires that no operations that may increase the reactivity of the core shall be made if CTS 3.8.A.4 is not met. ITS LCO

3.9.2 requires that two source range neutron flux monitors be operable in Mode 6. ITS LCO 3.9.2, Required Actions A.1 and A.2, require suspending Core Alterations and positive reactivity additions if only one SRM is Operable. Additionally, ITS LCO 3.9.2, Required Action B.1, requires immediate initiation of action to restore one SRM whenever both are inoperable. These are administrative changes with no impact on safety because . ITS LCO 3.9.2, Required Actions A.1, A.2 and B.1, are consistent with a reasonable interpretation of the existing requirements.

A.4 CTS 3.8.A specifies that requirements for source range monitors are applicable during handling operations and during reactor vessel head removal or installation; however, CTS 3.8.A.4 specifies that at least one source range neutron flux monitor shall be in service when core geometry is not being changed. ITS LCO 3.9.2, LCO and Applicability, requires 2 SRMs Operable in Mode 6; however, ITS LCO 3.9.2, Required Actions A.1 and A.2, require suspending Core Alterations and positive reactivity additions if only one SRM is Operable. These are administrative changes with no impact on safety because the combination of the ITS LCO 3.9.2 Applicability and Required Actions A.1 are consistent with a reasonable interpretation of the existing requirements.

MORE RESTRICTIVE

M.1 CTS 3.8.A.4 and CTS 3.8.B implicitly require suspending Core Alterations and positive reactivity additions if only one SRM is Operable and immediate initiation of action to restore at least one SRM whenever both are inoperable. However, CTS 3.8.A.4 and CTS 3.8.B do not require more frequent verification of the boron concentration of all filled portions of the RCS and the refueling canal if no SRMs are Operable. ITS LCO 3.9.2, Required Action B.2, is added to require performing SR 3.9.1.1 (i.e., verification of refueling boron concentration) once per 12 hours if two required source range neutron flux monitors are inoperable. This change is needed because SRMs, including the associated audible alarms, are assumed to be Operable for the detection of a boon dilution event in sufficient time for operator actions needed to prevent an unplanned criticality. More frequent performance of SR 3.9.2.1 has no adverse impact on safety.

M.2 CTS 3.8.A.4 requires that the subcritical core be continuously monitored by two SRMs whenever core geometry is being changed and by at least one SRM at other times in Mode 6; however, there are no surveillance requirements for the periodic calibration of SRMs during refueling. ITS SR 3.9.2.2 is added to require performance of an SRM channel calibration every 24 months. The 24 month Frequency is the same as the requirement for calibration of the SRM trip function which is required to be Operable during reactor plant startups. This change is needed because it ensures that the SRM detector plateau or pre-amp discriminator curves are consistent with manufacturers' data for SRM performance. This process ensures the Operability of the source range neutron flux monitors. This change is acceptable because it does not introduce any operation which is un-analyzed while requiring periodic verification of SRM Operability at the same frequency as when the instrument is used during plant startups. Therefore, this change has no adverse impact on safety.

M.3 CTS 3.8.A.4 specifies that only one source range monitor is required if core geometry is not being changed; therefore, only one SRM is required during a positive reactivity addition that does not involve a change to core geometry. ITS LCO 3.9.2 requires 2 SRMs Operable in Mode 6 and ITS LCO 3.9.2, Required Actions A.1 and A.2, require suspending Core Alterations and positive reactivity additions if only one SRM is Operable. Therefore, ITS LCO 3.9.2 requires 2 SRMs during a positive reactivity addition that does not involve a change to core geometry. This change is needed because it requires redundant reactivity change monitoring capability during positive reactivity additions when the reactor vessel head is removed. This change is acceptable because it does not introduce any operation which is un-analyzed while redundant reactivity change monitoring capability during positive reactivity additions when the reactor vessel head is removed. Therefore, this change has no adverse impact on safety.

LESS RESTRICTIVE

None

REMOVED DETAIL

LA.1 CTS 3.8.A.4 requires that the subcritical core be continuously monitored by two SRMs and that each of these SRMs have continuous visual indication in the control room whenever core geometry is being changed. ITS LCO 3.9.2 maintains the requirement for two SRMs in Mode 6 and that Core Alterations and positive reactivity additions must be terminated if only one SRM is Operable; however, the detail related to visual indication is not included in ITS LCO 3.9.2 and is relocated to the FSAR and ITS Bases. This change is acceptable because ITS LCO 3.9.2 still requires that SRMs are Operable and the requirements for Operability are defined in the ITS LCO 3.9.2 Bases.

This change is acceptable because ITS 3.9.2 maintains the existing requirement for the Operability of two SRM channels; therefore, there is no change to the existing requirements and no change to the level of safety of facility operation.

This change, which allows the description of the design of the SRMs to be maintained in the FSAR and the detailed description of the requirements for Operability of the SRMs to be maintained in the ITS Bases, is consistent with the approach used in NUREG-1431 for all Limiting Conditions for Operation (LCOs). This approach is acceptable because the requirements of 10 CFR 50.59, Changes, Tests and Experiments, and ITS 5.5.13, Technical Specifications (TS) Bases Control Program, are designed to assure that changes to the FSAR and ITS Bases do not result in changes to the Technical Specification requirements and do not result in significant increases in the probability or consequences of accidents previously evaluated, do not create the possibility of a new or different kind of accident, and do not result in a significant reduction in a margin of safety. Additionally, IP3 programs that implement FSAR changes in accordance with 10 CFR 50.59 and ITS Bases changes in accordance with ITS 5.5.13 require periodic submittal of FSAR and Bases changes to the NRC for review.

This change is a less restrictive administrative change with no impact on safety because no requirements are being deleted from Technical Specifications and an appropriate change control process and an appropriate level of regulatory oversight is maintained for the information being relocated out of the Technical Specifications.

LA.2 CTS 3.8.A.4 requires that the subcritical core be continuously monitored by two SRMs and that one SRM has audible indication in the containment available whenever core geometry is being changed. ITS LCO 3.9.1 maintains the requirement for two SRMs in Mode 6 and that Core Alterations and positive reactivity additions must be terminated if only one SRM is Operable; however, the detail related to audible indication in containment is not included in ITS LCO 3.9.2 and is relocated to the FSAR and will be implemented by plant procedures.

This change is acceptable because ITS 3.9.2 maintains the existing requirement for the Operability of two SRM channels (including visual indication and either alarms and/or audible indication in the control room). This ensures that redundant monitoring capability needed for the prompt identification of a boron dilution event is maintained in the Technical Specifications. However, audible SRM indication in the containment, which is intended to alert operators on the refueling floor of reactivity changes due to fuel mis-positioning, is not retained in Technical Specifications because ITS LCO 3.9.1, Boron Concentration, provides protection for this event.

This change is a less restrictive administrative change with no impact on safety because ITS 3.9.2 maintains the requirements to have audible SRM indication in the containment during changes in core geometry. Therefore, requirements to have audible SRM indication in the containment during changes in core geometry can be maintained in the TRM with no significant adverse impact on safety.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.9.3:
"Containment Penetrations"

PART 1:

Indian Point 3
Improved Technical Specifications and Bases

3.9 REFUELING OPERATIONS

3.9.3 Containment Penetrations

LCO 3.9.3 The containment penetrations shall be in the following status:

a. The equipment hatch closed and held in place by at least four bolts or the equipment hatch opening is closed using an equipment hatch closure plate that may include a closed personnel access door;

b. One door in each air lock closed;

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:

1. closed by a manual or automatic isolation valve, a blind flange, or equivalent, or

2. capable of being closed by OPERABLE Containment Purge Isolation System.

--------------------------NOTE-------------------------- RAI-5

LCO 3.9.3.d and LCO 3.9.3.e are not required to be met if the reactor has been subcritical for ≥ 550 hours.

----

d. The Containment Purge System flow path shall be either:

1. closed by a manual or automatic isolation valve, a blind flange, or equivalent, or

2. aligned to discharge through the HEPA filters and charcoal adsorbers.

e. The Containment Pressure Relief Line shall be closed by a manual or automatic isolation valve, a blind flange, or equivalent.

APPLICABILITY: During CORE ALTERATIONS,
During movement of irradiated fuel assemblies within containment.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.9.3:
"Containment Penetrations"

PART 3:

DISCUSSION OF CHANGES

Differences between CTS and ITS

aligned to discharge through the HEPA filters and charcoal adsorbers if fuel is being moved prior to a specified decay time; however, ITS SR 3.9.3.2 requires verification every 7 days that this requirement is met. This change is needed because it requires periodic verification that LCO 3.9.3.d is being met. This change is acceptable because it does not introduce any operation that is un-analyzed while requiring periodic verification that LCO requirements are met. Therefore, this change has no adverse impact on safety.

M.3 CTS 3.8.A.9 requires that both the Containment Vent (i.e., 10 -inch pressure relief line per FSAR 5.3.2.4) and Containment Purge System (i.e., 36-inch supply and exhaust lines per FSAR 5.3.2.3) must be aligned to discharge through HEPA filters and charcoal adsorbers during movement of fuel until the reactor has been shutdown for a specified minimum number of hours.

ITS LCO 3.9.3.d maintains the requirement that the Containment Purge System (i.e., 36-inch supply and exhaust lines) be aligned to discharge through HEPA filters and charcoal adsorbers during movement of fuel until the reactor has been shutdown for a specified minimum number of hours; however, ITS LCO 3.9.3.e requires that the Containment Vent (i.e., 10 -inch pressure relief line) must be isolated until the reactor has been shutdown for a specified minimum number of hours. This change eliminates the option of having the Containment Vent (i.e., 10 -inch pressure relief line) open during the early stages of refueling.

This change is needed because the specific requirement in ITS LCO 3.9.3.e to keep the Containment Vent (i.e., 10 -inch pressure relief line) isolated until the reactor has been shutdown for a specified minimum number of hours is the justification for relocating requirements for testing of the associated HEPA filters and charcoal adsorbers from the Technical Specification to the Technical Requirements Manual (TRM). This change is acceptable because it is a reasonable interpretation of the existing requirement.

M.4 CTS 3.8.A.2 requires that at least one isolation valve shall be operable, locked closed or blind flanged in each line penetrating the containment. ITS 3.9.3.c.1 maintains this requirement except that, as clarified in the Bases, isolation may be accomplished by an OPERABLE automatic isolation valve. Specifying the operable isolation valve to be automatic is a more restricitve change. This change is needed to

NYPA

provide assurance that this potential pathway for releases to the environment is closed in the event of a fuel handling accident.

LESS RESTRICTIVE

L.1 CTS 3.8.A.8 requires that the Containment Building Vent and Purge System, including radiation monitors that initiate isolation, must be tested and verified to be operable within 100 hours prior to refueling operations. ITS SR 3.9.3.3 maintains the requirement for periodic verification that each required containment purge and exhaust valve actuates to the isolation position on an actual or simulated actuation signal; however, the SR Frequency is extended from within 100 hours prior to refueling operations to once every 92 days. This change is needed because it eliminates an ambiguity that could be interpreted as requiring the performance this SR every time refueling activities are started and stopped during a single refueling outage. This change is acceptable because the requirement that the Containment Building Vent and Purge isolation function is Operable during fuel movement and Core Alterations is unchanged. Elimination of the requirement to perform this verification within 100 hours prior to an activity is not significant because the normal periodic Surveillance Frequency is established to provide adequate assurance that requirements are being met. The 92-day Frequency ensures that the SR is performed at the start of each refueling and this Frequency is sufficient to provide a high degree of assurance that the valves will function as required throughout a refueling outage. Therefore, this change has no adverse impact on safety. RAI-6

REMOVED DETAIL

None

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.9.3:
"Containment Penetrations"

PART 5:

NUREG-1431
Annotated to show differences between NUREG-1431 and ITS

Status of NUREG 1431 Generic Changes for ITS 3.9.3

This ITS Specification is based on NUREG-1431 Specification No. **3.9.4**
as modified by the following Generic Changes:

OG No.	TSTF No.	Generic Change Description	NRC STATUS	IP3 STATUS	JD No.
BWROG-017	051 R0	REVISE CONTAINMENT REQUIREMENTS DURING HANDLING IRRADIATED FUEL AND CORE ALTERATIONS (REQUIREMENTS LIMITED TO "RECENTLY" IRRADIATED FUEL)	NRC Review	Not Incorporated	N/A
CEOG-021	068 Rx	CONTAINMENT PERSONNEL AIRLOCK DOORS OPEN DURING FUEL MOVEMENT	See Next Rev.	Superceded-See Next Rev	N/A
CEOG-021 R1	068 R1	CONTAINMENT PERSONNEL AIRLOCK DOORS OPEN DURING FUEL MOVEMENT	NRC Review	Not Incorporated	N/A
CEOG-112	196 R0	REVISE ISOLATION DEVICES TO INCLUDE ASME/ANSI EQUIVALENT METHODS	Rejected by NRC	Not Incorporated	N/A

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.9.3:
"Containment Penetrations"

CEOG-115	**197 R1**	REQUIRE CONTAINMENT CLOSURE	NRC Review	Not Incorporated	**N/A**
WOG-018	**092 R1**	REVISE THE CONTAINMENT PURGE AND EXHAUST SR TO EXEMPT VALVES THAT ARE LOCKED, SEALED OR SECURED	NRC Review	Not Incorporated	**N/A**
WOG-076	**312 R0**	ADMINISTRATIVELY CONTROL CONTAINMENT PENETRATIONS	Rejected by TSTF	Not Incorporated	**N/A**
WOG-120	**286 R0**	DEFINE OPERATION INVOLVING POSITIVE REATIVITY ADDITION	TSTF Review	Not Incorporated	**N/A**

3.9 REFUELING OPERATIONS

3.9.~~6~~ 3 Containment Penetrations

<CTS> LCO 3.9.~~6~~ 3 The containment penetrations shall be in the following status:

<3.8.A.1> <DOC A.4> a. The equipment hatch closed and held in place by ~~[~~four~~]~~ bolts; Insert: 3.9-6-01 CLB.1

<3.8.A.1> b. One door in each air lock closed; ~~and~~

<3.8.A.2> c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:

1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or

2. capable of being closed by an OPERABLE Containment Purge ~~and Exhaust~~ Isolation System. R.1

Insert: 3.9-6-02

APPLICABILITY: <DOC A.3> <3.8.A> During CORE ALTERATIONS,
During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.8.B> <DOC A8> A. One or more containment penetrations not in required status.	A.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2 Suspend movement of irradiated fuel assemblies within containment.	Immediately

NUREG-1431 Markup Inserts
ITS SECTION 3.9.3 - CONTAINMENT PENETRATIONS

INSERT: 3.9-6-01 (CLB1)

or the equipment hatch opening is closed using an equipment hatch closure plate that may include a closed personnel access door

INSERT: 3.9-6-02

-----------------------------NOTE-----------------------------

<38A.9> LCO 3.9.3.d and LCO 3.9.3.e are not required to be met if the reactor has been subcritical for ≥ 550 hours.

<38A.9> <DOC A.5> d. The Containment Purge System flow path shall be either:

<Doc M.3> 1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or | R.1

<Doc A.6> 2. aligned to discharge through the HEPA filters and charcoal adsorbers.

<Doc M.2> e. The Containment Pressure Relief Line shall be closed by a manual or automatic isolation valve, a blind flange, or equivalent.

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
<DOC M.1> SR 3.9.~~4~~.1 3	Verify each required containment penetration is in the required status.	7 days
SR 3.9.~~4~~.~~2~~ 3.3 <3.8.A.8> <DOC L.1>	Verify each required containment purge ~~and exhaust~~ system valve actuates to the isolation position on an actual or simulated actuation signal.	~~[18]~~ months 92 days

(CLB.1)

Insert: 3.9-7-02

Insert: 3.9-7-01

NUREG-1431 Markup Inserts
ITS SECTION 3.9.3 - CONTAINMENT PENETRATIONS

INSERT: 3.9-7-01

SR 3.9.3.2 <DOC A.6> <DOC M2>	--------------NOTE---------------- Not required to be met if the reactor has been subcritical for ≥ 550 hours. ---------------------------------- Verify Containment Purge System is either: a. closed by a manual or automatic isolation valve, blind flange, or equivalent, or b. aligned to discharge through the HEPA filters and charcoal adsorbers.	7 days

INSERT: 3.9-7-02

SR 3.9.3.4 <DOC A6> <DOC A7>	----------------NOTE------------------ Not required to be met if the reactor has been subcritical for ≥ 550 hours. -------------------------------------- Perform required Containment Purge System filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP

B 3.9 REFUELING OPERATIONS

B 3.9.~~4~~ 3 Containment Penetrations

BASES

BACKGROUND

During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, a release of fission product radioactivity within containment will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, this is accomplished by maintaining containment OPERABLE as described in LCO 3.6.1, "Containment." In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. The LCO requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths are closed ~~or capable of being closed~~ [except for the OPERABLE Purge System Penetrations]. Since there is no potential for containment pressurization, the Appendix J leakage criteria and tests are not required.

The containment serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained well within the requirements of 10 CFR 100. Additionally, the containment provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the equipment hatch must be held in place by at least four bolts. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced.

[Insert: B 3.9-11-01]

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 unit operation in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of unit shutdown

~~(continued)~~

NUREG-1431 Markup Inserts
ITS SECTION 3.9.3 - CONTAINMENT PENETRATIONS

INSERT: B 3.9-11 - 01

DB.1

In lieu of maintaining the equipment hatch inplace for containment closure, a temporary closure device may be used ~~in lieu of the equipment hatch~~ to maintain containment closure during core alterations or during movement of irradiated fuel assemblies within containment. The temporary closure device may provide penetrations for temporary services or personnel access. The temporary closure device will be designed to withstand a seismic event and designed to a withstand a pressure which ensures containment closure during refueling operations. The closure device will provide the same level of protection as that of the equipment hatch for the fuel handling accident by restricting direct air flow from the containment to the environment.

R.1

BASES

BACKGROUND (continued)

when containment closure is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, containment closure is required; therefore, the door interlock mechanism may remain disabled, but one air lock door must always remain closed.

The requirements for containment penetration closure ensure that a release of fission product radioactivity within containment will be restricted from escaping to the environment. The closure restrictions are sufficient to restrict fission product radioactivity release from containment due to a fuel handling accident during refueling.

Insert: B3.9-12-01

~~The Containment Purge and Exhaust System includes two subsystems. The normal subsystem includes a 42 inch purge penetration and a 42 inch exhaust penetration. The second subsystem, a minipurge system, includes an 8 inch purge penetration and an 8 inch exhaust penetration. During MODES 1, 2, 3, and 4, the two valves in each of the normal purge and exhaust penetrations are secured in the closed position. The two valves in each of the two minipurge penetrations can be opened intermittently, but are closed automatically by the Engineered Safety Features Actuation System (ESFAS). Neither of the subsystems is subject to a Specification in MODE 5.~~

~~In MODE 6, large air exchanges are necessary to conduct refueling operations. The normal 42 inch purge system is used for this purpose, and all four valves are closed by the ESFAS in accordance with LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation."~~

~~[The minipurge system remains operational in MODE 6, and all four valves are also closed by the ESFAS.~~

~~or~~

~~The minipurge system is not used in MODE 6. All four 8 inch valves are secured in the closed position.]~~

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere

~~(continued)~~

NUREG-1431 Markup Inserts
ITS SECTION 3.9.3 - CONTAINMENT PENETRATIONS

INSERT: B 3.9-12-01 (ducts) (DB.1)

The Containment Purge System consists of ~~a~~ the 36-inch containment purge supply and exhaust ducts. The supply system includes roughing filters, heating coils, fan and a containment penetration with two butterfly valves for isolation. The exhaust system includes a containment penetration with two butterfly valves for isolation and can be aligned to discharge to the atmosphere through the plant vent either directly or through the Containment Purge Filter System (i.e., a filter bank with roughing, HEPA and charcoal filters). R.1

The Containment Purge System must be isolated when in Modes 1, 2, 3 or 4 in accordance with requirements established in LCO 3.6.3, Containment Isolation Valves. In Modes 5 and 6, the Containment Purge System may be used for containment ventilation. When open, the Containment Purge System isolation valves are capable of closing in response to the detection of high radiation levels in accordance with requirements established in LCO 3.3.6, Containment Purge and Pressure Relief Isolation Instrumentation (Ref. 1). Despite this isolation capability, the Containment Purge System must be aligned to discharge through the Containment Purge Filter System during CORE ALTERATIONS or movement of irradiated fuel until the reactor has been shutdown for a specified minimum number of hours.

INSERT: B 3.9-12-01 (continued)

DB.1

The Containment Pressure Relief Line (i.e., Containment Vent) consists of a single 10-inch containment vent line that is used to handle normal pressure changes in the Containment when in Modes 1, 2, 3 and 4 (Ref. 1). The Containment Pressure Relief Line is equipped with three quick-closing butterfly type isolation valves, one inside and two outside the containment which isolate automatically in accordance with requirements established in LCO 3.3.2, Engineered Safety Feature Actuation System (ESFAS) Instrumentation, and LCO 3.3.6, Containment Purge and Pressure Relief Isolation Instrumentation. Although the Containment Pressure Relief Line discharges to the atmosphere via the Containment Auxiliary Charcoal Filter System (i.e., a filter bank with roughing, HEPA and charcoal filters), the Containment Pressure Relief Line must remain isolated during CORE ALTERATIONS or movement of irradiated fuel until the reactor has been shutdown for a specified minimum number of hours. The Containment Pressure Relief Line must remain isolated because the Containment Auxiliary Charcoal Filter System is not required to be tested in accordance with Specification 5.5.10, Ventilation Filter Test Program.

BASES

BACKGROUND (continued)

must be isolated on at least one side. Isolation may be achieved by an OPERABLE automatic isolation valve, or by a manual isolation valve, blind flange, or equivalent. Equivalent isolation methods must be approved in accordance with 10 CFR 50.59 and may include use of a material that can provide a temporary, atmospheric pressure, ventilation barrier for the other containment penetrations during fuel movements ~~(Ref. 1)~~. PA.1

APPLICABLE SAFETY ANALYSES

During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 2). Fuel handling accidents, analyzed in Reference ~~3~~ 2, include dropping a single irradiated fuel assembly and handling tool or a heavy object onto other irradiated fuel assemblies. ~~The requirements of LCO 3.9.7, "Refueling Cavity Water Level," and the minimum decay time of 100 hours prior to CORE ALTERATIONS ensures~~ Insert: B 3.9-13-01 that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are well within the guideline values specified in 10 CFR 100. Standard Review Plan, Section 15.7.4, Rev. 1 (Ref. 3), defines "well within" 10 CFR 100 to be 25% or less of the 10 CFR 100 values. The acceptance limits for offsite radiation exposure will be 25% of 10 CFR 100 values or the NRC staff approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits).

Containment penetrations satisfy Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36. PA.1

LCO

This LCO limits the consequences of a fuel handling accident in containment by limiting the potential escape paths for fission product radioactivity released within containment. The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed except for the OPERABLE containment purge ~~and exhaust~~ System penetrations. For the OPERABLE containment purge ~~and exhaust~~ System penetrations, this LCO ensures that these penetrations are isolable by the Containment Purge and Exhaust Isolation ~~System~~ instrumentation. The OPERABILITY requirements for this LCO ensure that Insert: B 3.9-13-02 the automatic purge ~~and exhaust~~ System valves DB.

~~(continued)~~

INSERT: B 3.9-13-01

DB.1

The release of radioactivity from the containment following a fuel handling accident is limited by the following:

a) The requirements of LCO 3.9.6, "Refueling Cavity Water Level;"

b) The minimum decay time of 145 hours prior to CORE ALTERATIONS; and,

c) The requirements of this LCO to either isolate the Containment Purge System or align the system to discharge through the HEPA filters and charcoal adsorbers for a minimum of first 550 hours following the reactor shutdown.

This combination of requirements ensures

INSERT: B 3.9-13-02

DB.1
CLB.1

Additionally, the requirement to isolate the Containment Purge System or align the system to discharge through the HEPA filters and charcoal adsorbers for a minimum of the first 550 hours following the reactor shutdown is required to limit offsite radiation exposure to within required limits. The Containment Pressure Relief Line must remain isolated because the Containment Auxiliary Charcoal Filter System is not required to be tested in accordance with Specification 5.5.10, Ventilation Filter Test Program.

BASES

LCO (continued)

~~Closure times specified in the FSAR can be achieved and, therefore~~ meet the assumptions used in the safety analysis to ensure that releases through the valves ~~are~~ are filtered and can terminated, such that radiological doses are within the acceptance limit.

APPLICABILITY

The containment penetration requirements are applicable during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment because this is when there is a potential for a fuel handling accident. In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1. In MODES 5 and 6, when CORE ALTERATIONS or movement of irradiated fuel assemblies within containment are not being conducted, the potential for a fuel handling accident does not exist. Therefore, under these conditions no requirements are placed on containment penetration status.

ACTIONS

A.1 and A.2

If the containment equipment hatch, air locks, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere is not in the required status, including the Containment Purge ~~and Exhaust~~ Isolation ~~System~~ instrumentation not capable of automatic actuation when the purge and exhaust valves are open, the unit must be placed in a condition where the isolation function is not needed. This is accomplished by immediately suspending CORE ALTERATIONS and movement of irradiated fuel assemblies within containment. Performance of these actions shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.~~4~~3.1

This Surveillance demonstrates that each of the containment penetrations required to be in its closed position is in that position. The Surveillance on the open purge and exhaust valves will demonstrate that the valves are not blocked from closing. Also the Surveillance will

~~(continued)~~

SURVEILLANCE REQUIREMENTS

SR 3.9.~~1~~3.1 (continued)

demonstrate that each valve operator has motive power, which will ensure that each valve is capable of being closed by an OPERABLE automatic containment purge and exhaust isolation signal.

The Surveillance is performed every 7 days during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations. A surveillance before the start of refueling operations will provide two or three surveillance verifications during the applicable period for this LCO. As such, this Surveillance ensures that a postulated fuel handling accident that releases fission product radioactivity within the containment will not result in a release of fission product radioactivity to the environment.

Insert: B3.9-15-01

SR 3.9.~~1.2~~ 3.3

This Surveillance demonstrates that each containment purge and exhaust valve actuates to its isolation position on ~~manual initiation or on~~ an actual or simulated high radiation signal. The 18 month [92 days] Frequency maintains consistency with other similar ESFAS instrumentation and valve testing requirements. [Insert: B3.9-15-02] In LCO 3.3.6, the Containment Purge ~~and Exhaust~~ Isolation instrumentation requires a CHANNEL CHECK every 12 hours and a COT every 92 days to ensure the channel OPERABILITY during refueling operations. Every 18 [24] months a CHANNEL CALIBRATION is performed. The system actuation response time is demonstrated every 18 months, during refueling, on a STAGGERED TEST BASIS. SR 3.6.3.5 demonstrates that the isolation time of each valve is in accordance with the Inservice Testing Program requirements. These Surveillances performed during MODE 6 will ensure that the valves are capable of closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the containment.

Insert: B3.9-15-03

NUREG-1431 Markup Inserts
ITS SECTION 3.9.3 - CONTAINMENT PENETRATIONS

INSERT: B 3.9-15-01

SR 3.9.3.2

This SR requires periodic verification every 7 days that the Containment Building Purge System is either isolated or aligned to discharge through the HEPA filters and charcoal adsorbers. This SR is needed because it requires periodic verification that LCO 3.9.3.d is being met. A Note provides the allowance that this SR is not required to be performed or met if the reactor has been subcritical for ≥ 550 hours. These restrictions ensure that the offsite dose limit for a fuel handling accident of 75 rem to the thyroid at the exclusion area boundary (i.e., 25 percent of the 10 CFR Part 100 limit of 300 rem) is met by either filtering any release from the containment or by allowing a greater decay time before fuel handling activities are permitted.

INSERT: B 3.9-15-02

ensures that this SR is performed prior to this function being required and periodically thereafter.

INSERT: B 3.9-15-03

SR 3.9.3.4

This SR verifies that the required Containment Building Purge System testing is performed in accordance with Specification 5.5.10, Ventilation Filter Test Program (VFTP). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

BASES (continued)

REFERENCES	1.	FSAR, Section 5.3 ~~GPU Nuclear Safety Evaluation SE-0002000-001, Rev. 0, May 20, 1988.~~
	2.	FSAR, Section ~~15.4.5~~ 14.2.
	3.	NUREG-0800, Section 15.7.4, Rev. 1, July 1981.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.9.5:

"Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level"

PART 1:

Indian Point 3
Improved Technical Specifications and Bases

Rev 1 correction to 1 Bases page only
to properly reflect STS markup (Part 5)
in Rev 0.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.5.1 (continued)

water level in the vicinity of the reactor vessel nozzles, the RHR pump suction requirements must be met. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator for monitoring the RHR System in the control room.

SR 3.9.5.2

Verification that the required pump not in operation is OPERABLE ensures that an additional RCS or RHR pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pump. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

NYPA

REFERENCES

1. FSAR, Section 6.2.