# JAFNPP

## IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

## ITS: 5.5

### Programs and Manuals

MARKUP OF NUREG-1433, REVISION 1
SPECIFICATION

## 5.0 ADMINISTRATIVE CONTROLS

### 5.5 Programs and Manuals

The following programs and manuals shall be established, implemented and maintained. PA1

5.5.1 Offsite Dose Calculation Manual (ODCM)

[6.17.A] [1.0.H] a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program; and

[M6] b. The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Radioactive Effluent Release ~~reports~~ Reports required by Specification [5.6.2] and Specification [5.6.3]. PA1 PA2

c. Licensee initiated changes to the ODCM: PA3

[6.17.C.1.c] 1 ~~a.~~ Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

[6.17.C.1.a] (a) ~~1.~~ sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), and

[6.17.C.1.b] (b) ~~2.~~ a determination that the change(s) maintain the levels of radioactive effluent control required ~~by~~ pursuant to 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations; X1

[6.17.C.2] [M1] 2 ~~b.~~ Shall become effective after ~~review and acceptance by the onsite review function and~~ the approval of the [Plant ~~Superintendent~~] manager; and TA11 TA2

TSTF-76 R1

[6.17.C.1] 3 ~~c.~~ Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page

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(continued)

## 5.5 Programs and Manuals

5.5.1 Offsite Dose Calculation Manual (ODCM) (continued)

that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

[FOL 2.C.4] 5.5.2 Primary Coolant Sources Outside Containment

[M7] Reactor Water Cleanup, DB1

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include (the Low Pressure Core Spray, High Pressure Coolant Injection, Residual Heat Removal, Reactor Core Isolation Cooling, hydrogen recombiner, process sampling, and Standby Gas Treatment). The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

24 month PA4

TA12 b. Integrated leak test requirements for each system at refueling cycle intervals or less. X2

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the 24 month Frequency for performing integrated leak test activities. TSTF-299, RO

[L5]

[6.19] 5.5.3 Post Accident Sampling

This program provides controls that ensure the capability to obtain and analyze reactor coolant, radioactive gases, and particulates in plant gaseous effluents and containment atmosphere samples under accident conditions. The program shall include the following:

a. Training of personnel;

b. Procedures for sampling and analysis; and

c. Provisions for maintenance of sampling and analysis equipment.

[DOC A.6] 5.5.4 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably

(continued)

5.5 Programs and Manuals

[DOC A6] 5.5.4 Radioactive Effluent Controls Program (continued)

achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

TA5
INSERT 5.5.4-1
from the site
PA1
TSTF-258, R4
edit

b. Limitations on the concentrations of radioactive material released in liquid effluents ~~to unrestricted areas,~~ conforming to ~~10 CFR 20, Appendix B, Table 2, Column 2;~~

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;

the site
PA5

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from ~~each unit~~ to unrestricted areas, conforming to 10 CFR 50, Appendix I;

TA10
INSERT 5.5.4-3
TSTF-308, R1

e. ~~Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;~~

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

TA5
INSERT 5.5.4-2
from the site
PA1
at or
TSTF-258, R4

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary ~~conforming to the dose associated with 10 CFR 20, Appendix B, Table 2, Column 1;~~

(continued)

Insert 5.5.4-1 TA 5

10 times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001-20.2402;

TSTF-258, R4

Insert 5.5.4-2 TA5

shall be in accordance with the following:

1. For noble gases: a dose rate ≤ 500 mrem/yr to the whole body and a dose rate ≤ 3000 mrem/yr to the skin, and

2. For iodine-131, iodine-133, tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: a dose rate ≤ 1500 mrem/yr to any organ;

TSTF-258, R4

Insert 5.5.4-3 TA10

Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days.
Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days.

TSTF-308, R1

5.5 Programs and Manuals

[A6] 5.5.4 Radioactive Effluent Controls Program (continued)

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from ~~each unit~~ the site to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

PA5

PA1 edit

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives > at or 8 days in gaseous effluents released from ~~each unit~~ the site to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

PA5 TSTF-258, R4

TA5

j. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190; and

INSERT 5.5.4-4

TA5

k. Limitations on venting and purging of the ~~Mark II~~ primary containment through the Standby Gas Treatment System to maintain releases as low as reasonably achievable ~~(in BWR/4s with Mark II containments)~~.

DB3 DB2

PA6

[6.10.B.6] 5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the UFSAR Section [4.2], cyclic and transient occurrences to ensure that components are maintained within the design limits.

[6.20] 5.5.6 ~~Pre-Stressed Concrete Containment Tendon Surveillance Program~~

~~This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with [Regulatory Guide 1.35, Revision 3, 1989].~~

~~The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.~~

CLB1

INSERT 5.5.6-1

TA3 TSTF-52, R3

(continued)



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TSTF-258, R4

Insert 5.5.4-4 TA5

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

Insert 5.5.6-1 TA3 CLB1

5.5.6 Primary Containment Leakage Rate Testing Program

This program implements the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exception:

Type C testing of valves not isolable from the containment free air space may be accomplished by pressurization in the reverse direction, provided that testing in this manner provides equivalent or more conservative results than testing in the accident direction. If potential atmospheric leakage paths (e.g., valve stem packing) are not subjected to test pressure, the portions of the valve not exposed to test pressure shall be subjected to leakage rate measurement during regularly scheduled Type A testing. A list of these valves, the leakage rate measurement method, and the acceptance criteria, shall be contained in the Program.

a. The peak primary containment internal pressure for the design basis loss of coolant accident, $P_a$, is 45 psig.

b. The maximum allowable primary containment leakage rate, $L_a$, at $P_a$, shall be 1.5% of containment air weight per day.

c. The leakage rate acceptance criteria are:

   1. Primary containment leakage rate acceptance criteria is $\leq 1.0\ L_a$. During plant startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and Type C tests, and $\leq 0.75\ L_a$ for the Type A tests.

(continued)

Insert 5.5.6-1 (continued) TA3 CLB

2. Air lock testing acceptance criteria are:

   (a) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$; and

   (b) For each door seal, leakage rate is $\leq$ 120 scfd when tested at $\geq P_a$.

3. MSIV leakage rate acceptance criteria is $\leq$ 11.5 scfh for each MSIV when tested at $\geq$ 25 psig.

d. The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.

e. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

TSTF-52, R3

5.5 Programs and Manuals (continued)

CLB2

pumps and valves

certain

TSTF-279, R0

[4.0.E.1] 5.5.7 Inservice Testing Program

CLB2

[A8] This program provides ~~controls~~ for inservice testing of certain ASME Code Class 1, 2, and 3 ~~components~~ ~~including applicable supports~~. The program shall include the following:

PA7

PA1

[4.0.E.2] a. Testing ~~frequencies~~ specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda are as follows:

[1.0.T]

| ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice testing activities | Required Frequencies for performing inservice testing activities |
|---|---|
| Weekly | At least once per 7 days |
| Monthly | At least once per 31 days |
| Quarterly or every 3 months | At least once per 92 days |
| Semiannually or every 6 months | At least once per 184 days |
| Every 9 months | At least once per 276 days |
| Yearly or annually | At least once per 366 days |
| Biennially or every 2 years | At least once per 731 days |

[4.0.E.3] b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;

[Doc A8] c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

[4.0.E.5] d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.

Safeguards

[Doc M2] 5.5.8 Ventilation Filter Testing Program (VFTP)

PA6

This ~~A~~ program ~~shall be established to~~ implement S the following required testing of Engineered ~~Safety Features (ESF)~~ filter ventilation systems~~, at the frequencies specified in [Regulatory Guide ], and in accordance with [Regulatory Guide 1.52, Revision 2, ASME N510-1989, and AG-1]~~.

INSERT 5.5.8-1

CLB3

(continued)



PA7 The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test Frequencies.

REVISION H

Insert 5.5.8-1 CLB 3

The tests described in Specifications 5.5.8.a and 5.5.8.b shall be performed: RAI 5.5-3

Once per 24 months;

After each complete or partial replacement of the HEPA filter or charcoal adsorber filter; or after removal of a charcoal sample;

After any structural maintenance on the HEPA filter or charcoal adsorber housing that could affect the filter system efficiency; and RAI 5.5-2

Following painting, fire, or chemical release that could adversley affect the ability of the filter system to perform the intended function in any ventilation zone communicating with the system.

The tests described in Specification 5.5.8.c shall be performed:

Once per 24 months; RAI 5.5-3

After 720 hours of system operation;

After any structural maintenance on the charcoal adsorber housing that could affect the filter system efficiency; and

Following painting, fire, or chemical release that could adversley affect the ability of the charcoal filter system to perform the intended function in any ventilation zone communicating with the system.

The tests described in Specifications 5.5.8.d and 5.5.8.e shall be performed once per 24 months. RAI 5.5-2

PA6 Engineered Safeguards

RAI 5.5-4

## 5.5 Programs and Manuals

### 5.5.8 Ventilation Filter Testing Program (VFTP) (continued)

Sections C.5.a and C.5.c of

PA8

[4.7.B.1.b(1)]
[4.11.A.1.b]

a. Demonstrate for each of the ESF systems that an inplace test of the HEPA filters shows a penetration and system bypass < [0.05]% when tested in accordance with [Regulatory Guide 1.52, Revision 2, and ASME N510-1989] at the system flowrate specified below [± 10%].

1980 CLB9

indicated below

PA9

| ESF Ventilation System | Flowrate |
|---|---|
| [ ] | [ ] |

X6

PA9
INSERT 5.5.8-2

PA6
Engineered Safeguards

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < [0.05]% when tested in accordance with [Regulatory Guide 1.52, Revision 2, and ASME N510-1989] at the system flowrate specified below [± 10%].

CLB9
indicated below
1980

[4.7.B.1.b.(2)]
[4.11.A.1.c]

PA9 PA8
Sections C.5.a and C.5.d of

X6

| ESF Ventilation System | Flowrate |
|---|---|
| [ ] | [ ] |

TA8
INSERT 5.5.8-3 PA9

PA6
Engineered Safeguards

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in [Regulatory Guide 1.52, Revision 2], shows the methyl iodide penetration less than the value specified below when tested in accordance with [ASTM D3803-1989] at a temperature of ≤ [30°C] and greater than or equal to the relative humidity specified below.

Section C.6.b of

[4.7.B.1.c]
[4.11.A.2]

CLB7

(86°F) CLB7

TA8

| ESF Ventilation System | Penetration | RH |
|---|---|---|
| [ ] | [ ] | [ ] |

[M8]

INSERT 5.5.8-4 X6

CTS Amend 269

TSTF-362, R0

(continued)

Insert 5.5.8-2 PA 9 TA 8

| Engineered Safeguards Ventilation System | Penetration | Flowrate, scfm |
|---|---|---|
| Standby Gas Treatment System | < 1.0% | 5,400 to 6,600 |
| Control Room Emergency Ventilation Air Supply System | < 1.0% | 900 to 1,100 |

Insert 5.5.8-3 PA 9 TA 8

| Engineered Safeguards Ventilation System | Penetration | Flowrate, scfm |
|---|---|---|
| Standby Gas Treatment System | < 1.0% | 5,400 to 6,600 |
| Control Room Emergency Ventilation Air Supply System | < 0.5% | 900 to 1,100 |

Insert 5.5.8-4 X6 TA 8

| Engineered Safeguards Ventilation System | Penetration | RH |
|---|---|---|
| Standby Gas Treatment System | < 5% | ≥ 70% |
| Control Room Emergency Ventilation Air Supply System | < 5% | ≥ 95% |

CTS Amend 269

5.5 Programs and Manuals

5.5.8 Ventilation Filter Testing Program (VFTP) (continued)

Editorial

X3

[Reviewer's Note: Allowable penetration = [100% - methyl iodide efficiency for charcoal credited in staff safety evaluation]/(safety factor).

Safety factor = [5] for systems with heaters.
= [7] for systems without heaters.]

[4.11.A.1.a]
[4.7.B.1.a.1]

PA6 Engineered Safeguards

d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is less than the value specified below when tested in accordance with [Regulatory Guide 1.52, Revision 2, and ASME N510-1989] at the system flowrate specified as follows [± 10%]: PA9

CLB8 X4 PA9

INSERT 5.5.8-5

| ESF Ventilation System | Delta P | Flowrate |
|---|---|---|
| [ ] | [ ] | [ ] |

Standby Gas Treatment

PA1

[4.7.B.1.a.2]

[e. Demonstrate that the heaters for each of the ESF system dissipate the value specified below [± 10%] when tested in accordance with [ASME N510-1989]: 75 X6 > 29 kW

| ESF Ventilation System | Wattage |
|---|---|
| [ ] | [ ] |
]

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies. PA7

CTS Amend 170

[Doc A10] 5.5.9 Explosive Gas and Storage Tank Radioactivity Monitoring Program

DB4

[6.22] This program provides controls for potentially explosive gas mixtures contained in the [Waste Gas Holdup System], [the quantity of radioactivity contained in gas storage tanks or fed into the offgas treatment system, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks]. The

Main Condenser Offgas Treatment System

PA6

(continued)

Insert 5.5.8-5

| Engineered Safeguards Ventilation System | Delta P, inches wg | Flowrate, scfm |
|---|---|---|
| Standby Gas Treatment System | 5.7 | 5,400 to 6,600 |
| Control Room Emergency Ventilation Air Supply System | 5.8 | 900 to 1,100 |

5.5 Programs and Manuals

[A10] 5.5.9 Explosive Gas and Storage Tank Radioactivity Monitoring Program (continued)

DB4

gaseous radioactivity quantities shall be determined following the methodology in [Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure"]. The liquid radwaste quantities shall be determined in accordance with [Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures"].

CLB 4

The program shall include:

Main Condenser Offgas Treatment

PA6

CTS Amend 270

[6.22]

a. The limits for concentrations of hydrogen and oxygen in the ~~[Waste Gas Holdup]~~ System[s] and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion);

b. ~~A surveillance program to ensure that the quantity of radioactivity contained in [each gas storage tank and fed into the offgas treatment system] is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of [an uncontrolled release of the tanks' contents];~~ and

DB4

[RETS 2.5a]
[A10] DB4
[RETS 2.5] PA6
X8

b. ~~c.~~ A surveillance program to ensure that the quantity of radioactivity contained in all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the [Liquid Radwaste ~~(Treatment)~~ System] is less than the amount that would result in ~~concentrations less than the limits of~~ ~~10 CFR 20~~ Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply ~~in~~ ~~an unrestricted area~~, in the event of an uncontrolled release of the tanks' contents.

a concentration that is 10 times the concentration values in

RAI 5.5-7

[L6]

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program Surveillance Frequencies.

PA1

to 10 CFR 20.1001 - 20.2402 (excluding Tritium and dissolved or entrained noble gases)

CLB 5

beyond the site boundary

PA 6

(continued)



REVISION H

5.5 Programs and Manuals (continued)

[Doc M3] 5.5.10 Diesel Fuel Oil Testing Program

This ~~A diesel fuel oil testing~~ program ~~to~~ implements required testing of both new fuel oil and stored fuel oil ~~shall be established~~. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following: PA1

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

PA1

1. ~~an~~ API gravity or an absolute specific gravity within limits,

2. ~~a~~ flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and

3. ~~a~~ clear and bright appearance with proper color;

Editorial

b. ~~Other properties for ASTM 2D fuel oil are within limits~~ within 31 days following ~~sampling and~~ addition of the new fuel oil to storage tanks; and TA1

INSERT 5.5.10-1

c. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every 31 days in accordance with ASTM D-~~2276~~ 5452-1996 ~~Method A-2 or A-3~~. X7

INSERT 5.5.10-2

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program test Frequencies. TA4

[Doc M4] 5.5.11 X7 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews. TA9

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not ~~involve~~ require either of the following: PA6

TSTF-364, R0

PA1

1. ~~a~~ change in the TS incorporated in the license; or

2. ~~a~~ change to the ~~updated FSAR~~ UFSAR or Bases that ~~involves an unreviewed safety question as defined in~~ requires NRC approval persuant to 10 CFR 50.59. TA9

(continued)



REVISION H

Insert 5.5.10-1 TA 1

verify that the properties of the new fuel oil, other than those addressed in a., above, are within limits for ASTM 2D fuel oil

Insert 5.5.10-2 X7

except that the 0.8 micron filters specified in the ASTM may be replaced with membrane filters up to 3.0 microns

## 5.5 Programs and Manuals

[Doc M4] 5.5.11 Technical Specifications (TS) Bases Control Program (continued)

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the ~~FSAR~~ UFSAR.

PA6

d. Proposed changes that meet the criteria of Specification 5.5.11b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

[Doc M5] 5.5.12 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate limitations and remedial or compensatory actions may be identified to be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a.

1 ~~a.~~ Provisions for cross division checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

2 ~~b.~~ Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

3 ~~c.~~ Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

4 ~~d.~~ Other appropriate limitations and remedial or compensatory actions.

PA3

TA6

b. A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

Insert 5.5.12-1

1 ~~a.~~ A required system redundant to system(s) supported by the inoperable support system is also inoperable; or

TSTF-273, R2
W06-ED-23

(continued)

Insert 5.5.12-1 TA6

no concurrent loss of offsite power or no concurrent loss of onsite diesel generator subsystems,

TSTF-273, R2

5.5 Programs and Manuals

[DocMS] 5.5.12 Safety Function Determination Program (SFDP) (continued)

(2) ~~b.~~ A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

(3) ~~c.~~ A required system redundant to support system(s) for the supported systems (~~a~~ 1) and (~~b~~ 2) above is also inoperable.

PA3

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

TSTF-273, R2

Insert 5.5.13 CLB6

Insert 5.5.12-2 TA6

Insert 5.5.12-2 TA6

When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

TSTF-273, R2

Insert 5.5.13 CLB6

CONFIGURATION RISK MANAGEMENT PROGRAM (CRMP)

The CRMP provides a proceduralized risk-informed assessment to manage the risk associated with equipment inoperability. The program applies to technical specification structures, systems, or components for which a risk-informed allowed outage time has been granted. The program is to include the following:

a. Provisions for the control and implementation of a Level 1 at-power internal events PRA-informed methodology. The assessment is to be capable of evaluating the applicable plant configuration.

b. Provisions for performing an assessment prior to entering the plant configuration described by the Limiting Condition for Operation (LCO) Condition(s) for preplanned activities.

c. Provisions for performing an assessment after entering the plant configuration described by the LCO Action Statement for unplanned entry into the LCO Condition(s).

d. Provisions for assessing the need for additional actions after the discovery of additional equipment-out-of-service conditions while in the plant configuration described by the LCO Condition(s).

e. Provisions for considering other applicable risk-significant contributors such as Level 2 issues and external events, qualitatively or quantitatively.

# JAFNPP

## IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

## ITS: 5.5

Programs and Manuals

JUSTIFICATION FOR DIFFERENCES (JFDs)
FROM NUREG-1433, REVISION 1

## JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
## ITS: 5.5 - PROGRAMS AND MANUALS

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 ITS 5.5.6 has been added to reflect JAFNPP Amendment No. 234 (October 4, 1996), which permitted implementation of 10 CFR part 50, Appendix J, Option B, as modified by approved exemptions, via the Primary Containment Leakage Rate Testing Program. These changes are consistent with TSTF-52, Revision 3. ITS 5.5.6, Primary Containment Leakage Rate Testing Program, replaces ISTS 5.5.6, Pre-stressed Concrete Containment Tendon Surveillance Program, which was deleted (DB2).

RAI 5.5-1

CLB2 ITS 5.5.7 is modified to state that the IST Program provides controls for "certain ASME Code Class 1, 2, and 3 pumps and valves" as required for plants whose construction permit was issued prior to January 1, 1971 in place of "ASME Code Class 1, 2, and 3 components including applicable supports." 10 CFR 50.55a(f) provides the regulatory requirements for an Inservice Testing (IST) Program, and specifies that ASME Code Class 1, 2, and 3 pumps and valves are those components covered by an IST Program. 10 CFR 50.55a(g) provides regulatory requirements for an Inservice Inspection (ISI) Program, and specifies that ASME Code Class 1, 2, and 3 components (including supports) are covered by the ISI Program, and that pumps and valves are covered by the IST Program in 10 CFR 50.55a(f). The ISTS does not include ISI Program requirements, as these program requirements have been relocated to plant specific documents. Therefore, the "applicable support" requirements are deleted and the components the IST Program applies to (i.e., pumps and valves) are added for clarity, consistent with current licensing basis. Deletion of the reference to "applicable support" requirements is also consistent with TSTF-279, R0)

TSTF-279, R0

CLB3 The Frequencies adopted in ITS 5.5.8, for performance of tests in the Ventilation Filter Testing Program (VFTP), are consistent with those specified in Regulatory Guide 1.52, Revision 2, except for the 24-month Frequencies, which support the change to a 24-month operating cycle.

CLB4 ITS 5.5.9 contains statements that specify the methodology to be used for determining quantities of radioactivity present in liquid radwaste holdup tanks. Consistent with current licensing basis, such methodology is contained in the ODCM, and need not be repeated in the ITS.

CLB5 ITS 5.5.9 is revised to reflect the current JAFNPP requirement in CTS RETS 2.5.a of less than or equal to 10 curies, excluding Tritium and dissolved or entrained noble gases, for quantity of liquid radioactive material in outside storage tanks.

CTS Amend 269

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB6 ITS 5.5.13 has been added to reflect License Amendment 253, dated July 30, 1999 (TAC No. M94611) which added CTS 6.21, Configuration Risk Management Program (CRMP).

CLB7 ITS 5.5.8.c has been revised to reflect the current JAFNPP requirement in CTS 4.7.B.1.c and CTS 4.11.A.2 that charcoal absorber samples are obtained as described in Section C.6.b of Regulatory Guide 1.52, Revision 2 and that the minimum temperature for analysis is 30°C or 86°F.

edit

CLB8 ITS 5.5.8.d has been revised to reflect the current JAFNPP licensing basis that in CTS 4.7.B.1.a.(1) and CTS 4.11.A.1.a, filter pressure drop testing does not include the system prefilters.

RAI 5.5-4

CLB9 ITS 5.5.8.a and ITS 5.5.8.b have been revised to reflect the current licensing basis. The HEPA filters and charcoal adsorbers in the Standby Gas Treatment (SGT) and Control Room Emergency Ventilation Air Supply (CREVAS) Systems were designed and constructed to ANSI/ASME N509-1976, Nuclear Power Plant Air Cleaning Units and Components. ASME N510-1980 is the appropriate standard for testing the SGT and CREVAS System HEPA filters and charcoal adsorbers.

PLANT SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

PA1 Editorial changes have been made for enhanced clarity or to correct a grammatical/typographical error.

PA2 The brackets have been removed and the proper ITS Specification numbers have been provided.

PA3 ITS 5.5.1, "Offsite Dose Calculation Manual," and 5.5.12, "Safety Function Determination Program," presentation format has been renumbered consistent with the Writer's Guide for the Restructured Technical Specifications.

PA4 ITS 5.5.2.b interval of refueling has been changed to 24 months for consistency with Surveillance Frequencies and the current refueling outage cycle length.

PA5 ITS 5.5 is modified to reflect that JAFNPP is a single unit plant site.

PA6 Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific system/structure/component nomenclature, equipment identification or description.

## JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
## ITS: 5.5 - PROGRAMS AND MANUALS

### PLANT SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

PA7 ITS 5.5.8 statement of applicability for SR 3.0.2 and SR 3.0.3 is moved from the end of Section 5.5.8, to be included with the program statement at the beginning of Section 5.5.8, for clarity.

PA8 ITS 5.5.8 is modified to reflect the specific sections of Regulatory Guide 1.52, Revision 2, which are applicable to the VFTP, for clarity.

PA9 ITS 5.5.8 bracketed 10% allowance for system flowrate is deleted and the specific system values are provided in the appropriate column.

PA10 Not Used.

TSTF-76, R1

### PLANT SPECIFIC DIFFERENCE IN DESIGN OR DESIGN BASIS (DB)

DB1 ITS 5.5.2 has been revised to reflect the specific JAFNPP design for Primary Coolant Sources Outside Containment, and identify the applicable systems (M7).

DB2 ISTS 5.5.6, Pre-stressed Concrete Containment Tendon Surveillance Program, has been deleted to reflect that JAFNPP is a BWR/4, with a Mark I containment and not subject to these requirements. In addition, other references to the Mark II containment are deleted.

### PLANT SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

DB3 ITS 5.5.5, has been revised to reflect the specific JAFNPP design requirements as referenced in UFSAR Section 4.2.

DB4 ITS 5.5.9, has been revised to reflect the specific JAFNPP design requirement, that JAFNPP is a BWR. As such, references to gas storage tanks, applicable only to PWRs, are deleted. Subsequent Section numbers are revised as required.

### DIFFERENCE BASED ON APPROVED TRAVELER (TA)

TA1 The changes presented in TSTF Technical Specification Change Traveler number 106, Revision 1, have been incorporated into the revised Improved Technical Specifications.

TA2 The changes presented in Technical Specification Task Force (TSTF) Technical Specification Change Travelers, number 65, Revision 1 and number 76, Revision 1, have been incorporated into the revised Improved Technical Specifications.

DIFFERENCE BASED ON APPROVED TRAVELER (TA)

TA3 The changes presented in TSTF Technical Specification Change Traveler number 52, Revision 3, have been incorporated into the revised Improved Technical Specifications.

TA4 The changes presented in TSTF Technical Specification Change Traveler number 118, Revision 0, have been incorporated into the revised Improved Technical Specifications.

TA5 The changes presented in TSTF Technical Specification Change Traveler number 258, Revision 4, have been incorporated into the revised Improved Technical Specifications.

TA6 The changes presented in TSTF Technical Specification Change Traveler number 273, Revision 2 (including WOG-ED-23), have been incorporated into the revised Improved Technical Specifications.

TA7 The changes presented in TSTF Technical Specification Change Traveler number 279, Revision 0, have been incorporated into the revised Improved Technical Specifications.

TA8 The changes presented in TSTF Technical Specification Change Traveler number 362, Revision 0 (including WOG-ED-27), have been incorporated into the revised Improved Technical Specifications.

TA9 The changes presented in TSTF Technical Specification Change Traveler number 364, Revision 0 (including WOG-ED-24), have been incorporated into the revised Improved Technical Specifications.

TA10 The changes presented in TSTF Technical Specification Change Traveler number 308, Revision 1, have been incorporated into the revised Improved Technical Specifications.

TA11 The changes presented in TSTF Technical Specification Change Traveler number 65, Revision 1, have been incorporated into the revised Improved Technical Specifications.

TA12 The changes presented in TSTF Technical Specification Change Traveler number 299, Revision 0, have been incorporated into the revised Improved Technical Specifications.

TSTFs as stated

DIFFERENCE BASED ON PENDING TRAVELER (TP)

None

TSTF-258, R4
RAI 5.5-1

DIFFERENCE FOR OTHER REASONS THAN ABOVE (X)

X1 Not Used.

X2 ITS 5.5.2 includes a statement of applicability (L5) of ITS SR 3.0.2 and SR 3.0.3 to clarify that the allowances for the 24 month Surveillance Frequency extensions do apply, since these SRs are not normally applied to Frequencies identified in Administrative Controls chapters of the Technical Specifications. This change is a clarification needed to maintain provisions that would be allowed in the LCO sections of the Technical Specifications, is consistent with TSTF-299, Revision 0 (JFD TA12), and is consistent with statements provided in other programs.

TSTF-299, R0

X3 The bracketed "Reviewer's Note" has been deleted. This information is for the NRC reviewer to understand exactly what is needed to meet this requirement. This is not meant to be retained in the final version of the plant specific information.

X4 ITS 5.5.8.d references to Regulatory Guide 1.52, Revision 2, and ASME N510-1989 are deleted, since no actual reference for pressure drop testing exists in the documents.

TSTF-118, R0

X5 Not Used.

X6 The bracketed values for penetration and system bypass have been incorporated into ITS 5.5.8 consistent with the current requirements in CTS 4.7.B.1 and 4.11.A.1 as modified by M2.

X7 ITS 5.5.10.c (M3) is revised to reflect the current JAFNPP fuel oil particulate concentration testing standard ASTM D5452-1996 Method A except that in place of the 0.8 micron filter membrane specified in ASTM D5452-1996, filters up to 3.0 microns may be used. The ASTM standard was originally intended for test of aircraft fuel rather than diesel fuel and the Emergency Diesel Generator (EDG) engine mounted filters for the engine and motor driven fuel pumps are 10 micron filters. Performing the fuel particulate test using a 3 micron filter membrane is conservative and therefore acceptable.

Editorial

X8 ITS 5.5.9.b is revised by discussing the radioactivity contained in unprotected outside liquid storage tanks in terms of the limits associated with uncontrolled release of the contents. The changes makes the phrases and words consistent with those contained in ITS 5.5.4.b (as modified by TSTF-258, R4)

RAI 5.5-7

# JAFNPP

## IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

## ITS: 5.5

Programs and Manuals

RETYPED PROPOSED IMPROVED TECHNICAL SPECIFICATIONS (ITS)

## 5.0 ADMINISTRATIVE CONTROLS

## 5.5 Programs and Manuals

The following programs and manuals shall be established, implemented and maintained.

5.5.1 Offsite Dose Calculation Manual (ODCM)

a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program; and

b. The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Radioactive Effluent Release Reports required by Specification 5.6.2 and Specification 5.6.3.

c. Licensee initiated changes to the ODCM:

1. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

   (a) sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), and

   (b) a determination that the change(s) maintain the levels of radioactive effluent control required pursuant to 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

TSTF-76, R1

2. Shall become effective after approval of the plant manager; and

3. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by

## 5.5 Programs and Manuals

5.5.1 Offsite Dose Calculation Manual (ODCM) (continued)

markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include the Core Spray, High Pressure Coolant Injection, Residual Heat Removal, Reactor Core Isolation Cooling, Reactor Water Cleanup, process sampling, and Standby Gas Treatment. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system at 24 month intervals or less.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the 24 month Frequency for performing integrated leak test activities.

5.5.3 Post Accident Sampling

This program provides controls that ensure the capability to obtain and analyze reactor coolant, radioactive gases, and particulates in plant gaseous effluents and containment atmosphere samples under accident conditions. The program shall include the following:

a. Training of personnel;

b. Procedures for sampling and analysis; and

c. Provisions for maintenance of sampling and analysis equipment.

(continued)

5.5 Programs and Manuals (continued)

5.5.4 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents from the site to unrestricted areas, conforming to 10 times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001-20.2402;

TSTF-258, R4

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;

ed. 4

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from the site to unrestricted areas, conforming to 10 CFR 50, Appendix I;

e. Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days. Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days.

TSTF-308, R1

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

(continued)

## 5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the site boundary shall be in accordance with the following:

1. For noble gases: a dose rate ≤ 500 mrem/yr to the whole body and a dose rate ≤ 3000 mrem/yr to the skin, and

2. For iodine-131, iodine-133, tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: a dose rate ≤ 1500 mrem/yr to any organ;

TSTF-258, R4

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from the site to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives > 8 days in gaseous effluents released from the site to areas at or beyond the site boundary, conforming to 10 CFR 50, Appendix I;

Ed.+

j. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190; and

k. Limitations on venting and purging of the primary containment through the Standby Gas Treatment System to maintain releases as low as reasonably achievable.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program Surveillance Frequency.

TSTF-258, R4

5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the UFSAR Section 4.2, cyclic and transient occurrences to ensure that components are maintained within the design limits.

(continued)

5.5 Programs and Manuals (continued)

5.5.6 Primary Containment Leakage Rate Testing Program

This program implements the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exception:

Type C testing of valves not isolable from the containment free air space may be accomplished by pressurization in the reverse direction, provided that testing in this manner provides equivalent or more conservative results than testing in the accident direction. If potential atmospheric leakage paths (e.g., valve stem packing) are not subjected to test pressure, the portions of the valve not exposed to test pressure shall be subjected to leakage rate measurement during regularly scheduled Type A testing. A list of these valves, the leakage rate measurement method, and the acceptance criteria, shall be contained in the Program.

a. The peak primary containment internal pressure for the design basis loss of coolant accident, $P_a$, is 45 psig.

b. The maximum allowable primary containment leakage rate, $L_a$, at $P_a$, shall be 1.5% of containment air weight per day.

c. The leakage rate acceptance criteria are:

 1. Primary containment leakage rate acceptance criteria is $\leq 1.0\ L_a$. During plant startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and Type C tests, and $\leq 0.75\ L_a$ for the Type A tests.

 2. Air lock testing acceptance criteria are:

  (a) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$; and

  (b) For each door seal, leakage rate is $\leq$ 120 scfd when tested at $\geq P_a$.

 3. MSIV leakage rate acceptance criteria is $\leq$ 11.5 scfh for each MSIV when tested at $\geq$ 25 psig.

(continued)

## 5.5 Programs and Manuals

5.5.6 Primary Containment Leakage Rate Testing Program (continued)

d. The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.

e. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

TSTF-52, R3

5.5.7 Inservice Testing Program

This program provides controls for inservice testing of certain ASME Code Class 1, 2, and 3 pumps and valves. The program shall include the following:

a. Testing Frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda are as follows:

| ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice testing activities | Required Frequencies for performing inservice testing activities |
|---|---|
| Weekly | At least once per 7 days |
| Monthly | At least once per 31 days |
| Quarterly or every 3 months | At least once per 92 days |
| Semiannually or every 6 months | At least once per 184 days |
| Every 9 months | At least once per 276 days |
| Yearly or annually | At least once per 366 days |
| Biennially or every 2 years | At least once per 731 days |

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

(continued)

5.5 Programs and Manuals (continued)

5.5.7 Inservice Testing Program (continued)

d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.

5.5.8 Ventilation Filter Testing Program (VFTP)

This program implements the following required testing of Engineered Safeguards filter ventilation systems.

The tests described in Specifications 5.5.8.a and 5.5.8.b shall be performed:

RAI 5.5-3

Once per 24 months;

After each complete or partial replacement of the HEPA filter train or charcoal adsorber filter or after removal of a charcoal sample;

After any structural maintenance on the HEPA filter or charcoal adsorber housing that could affect the filter system efficiency; and

RAI 5.5-2

Following painting, fire, or chemical release that could adversely affect the ability of the filter system to perform the intended function in any ventilation zone communicating with the system.

The tests described in Specification 5.5.8.c shall be performed:

Once per 24 months;

After 720 hours of system operation;

After any structural maintenance on the charcoal adsorber housing that could affect the filter system efficiency; and

Following painting, fire, or chemical release that could adversely affect the ability of the charcoal filter system to perform the intended function in any ventilation zone communicating with the system.

The tests described in Specifications 5.5.8.d and 5.5.8.e shall be performed once per 24 months.

(continued)

5.5 Programs and Manuals (continued)

5.5.8 Ventilation Filter Testing Program (VFTP) (continued)

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test Frequencies.

a. Demonstrate for each of the Engineered Safeguards systems that an inplace test of the HEPA filters shows a system bypass indicated below when tested in accordance with Sections C.5.a and C.5.c of Regulatory Guide 1.52, Revision 2, and ASME N510-1980 at the system flowrate specified below.

| Engineered Safeguards Ventilation System | | Flowrate, scfm |
|---|---|---|
| Standby Gas Treatment System | < 1.0% | 5,400 to 6,600 |
| Control Room Emergency Ventilation Air Supply System | < 1.0% | 900 to 1,100 |

RAI 5.5-4

b. Demonstrate for each of the Engineered Safeguards systems that an inplace test of the charcoal adsorber shows system bypass indicated below when tested in accordance with Sections C.5.a and C.5.d of Regulatory Guide 1.52, Revision 2, and ASME N510-1980 at the system flowrate specified below.

| Engineered Safeguards Ventilation System | | Flowrate, scfm |
|---|---|---|
| Standby Gas Treatment System | < 1.0% | 5,400 to 6,600 |
| Control Room Emergency Ventilation Air Supply System | < 0.5% | 900 to 1,100 |

(continued)

## 5.5 Programs and Manuals

### 5.5.8 Ventilation Filter Testing Program (VFTP) (continued)

c. Demonstrate for each of the Engineered Safeguards systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Section C.6.b of Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of ≤ 30°C (86°F) and the relative humidity specified below. Edit

| Engineered Safeguards Ventilation System | Penetration | RH |
|---|---|---|
| Standby Gas Treatment System | < 5% | ≥ 70% |
| Control Room Emergency Ventilation Air Supply System | < 5% | ≥ 95% |

CTS Amend 269

d. Demonstrate for each of the Engineered Safeguards systems that the pressure drop across the combined HEPA filters and the charcoal adsorbers is less than the value specified below when tested at the system flowrate specified as follows: Edit

| Engineered Safeguards Ventilation System | Delta P, inches wg | Flowrate, scfm |
|---|---|---|
| Standby Gas Treatment System | 5.7 | 5,400 to 6,600 |
| Control Room Emergency Ventilation Air Supply System | 5.8 | 900 to 1,100 |

(continued)

## 5.5 Programs and Manuals

5.5.8 <u>Ventilation Filter Testing Program (VFTP)</u> (continued)

e. Demonstrate that the heaters for the Standby Gas Treatment System dissipate > 29 kW when tested in accordance with ASME N510-1975.

5.5.9 <u>Explosive Gas and Storage Tank Radioactivity Monitoring Program</u>

This program provides controls for potentially explosive gas mixtures contained in the Main Condenser Offgas Treatment System, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks.

The program shall include:

a. The limits for concentrations of hydrogen and oxygen in the Main Condenser Offgas Treatment System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion);

b. A surveillance program to ensure that the quantity of radioactivity contained in all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste System is less than the amount that would result in a concentration that is 10 times the concentration values in Appendix B, Table 2, Column 2, to 10 CFR 20.1001 – 20.2402 (excluding Tritium and dissolved or entrained noble gases) at the nearest potable water supply and the nearest surface water supply beyond the site boundary, in the event of an uncontrolled release of the tanks' contents.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program Surveillance Frequencies.

5.5.10 <u>Diesel Fuel Oil Testing Program</u>

This program implements required testing of both new fuel oil and stored fuel oil. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with

(continued)

## 5.5 Programs and Manuals

5.5.10 Diesel Fuel Oil Testing Program (continued)

applicable ASTM Standards. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

   1. An API gravity or an absolute specific gravity within limits,

   2. A flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and

   3. A clear and bright appearance with proper color.

b. Within 31 days following addition of the new fuel oil to storage tanks verify that the properties of the new fuel oil, other than those addressed in a., above, are within limits for ASTM 2D fuel oil; and

c. Total particulate concentration of the fuel oil is $\leq$ 10 mg/l when tested every 31 days in accordance with ASTM D-5452-1996, Method A except that the 0.8 micron filters specified in the ASTM may be replaced with membrane filters up to 3.0 microns.

Edit

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program test Frequencies.

5.5.11 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

   1. A change in the TS incorporated in the license; or

   2. A change to the UFSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

TSTF-364, R0

5.5 Programs and Manuals

5.5.11 Technical Specifications (TS) Bases Control Program (continued)

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

d. Proposed changes that meet the criteria of Specification 5.5.11b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.12 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate limitations and remedial or compensatory actions may be identified to be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6.

a. The SFDP shall contain the following:

1. Provisions for cross division checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

2. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

3. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

4. Other appropriate limitations and remedial or compensatory actions.

TSTF-273, R2
WOG-ED-23

b. A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power or no concurrent loss of onsite diesel generator subsystems, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

## 5.5 Programs and Manuals

5.5.12 Safety Function Determination Program (SFDP) (continued)

1. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or

2. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

3. A required system redundant to support system(s) for the supported systems (1) and (2) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the supported system.

TSTF-273, R2

5.5.13 CONFIGURATION RISK MANAGEMENT PROGRAM (CRMP)

The CRMP provides a proceduralized risk-informed assessment to manage the risk associated with equipment inoperability. The program applies to technical specification structures, systems, or components for which a risk-informed allowed outage time has been granted. The program is to include the following:

a. Provisions for the control and implementation of a Level 1 at-power internal events PRA-informed methodology. The assessment is to be capable of evaluating the applicable plant configuration.

b. Provisions for performing an assessment prior to entering the plant configuration described by the Limiting Condition for Operation (LCO) Condition(s) for preplanned activities.

c. Provisions for performing an assessment after entering the plant configuration described by the LCO Action Statement for unplanned entry into the LCO Condition(s).

d. Provisions for assessing the need for additional actions after the discovery of additional equipment-out-of-service conditions while in the plant configuration described by the LCO Condition(s).

e. Provisions for considering other applicable risk-significant contributors such as Level 2 issues and external events, qualitatively or quantitatively.

# JAFNPP

## IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

## ITS: 5.6

Reporting Requirements

MARKUP OF CURRENT TECHNICAL SPECIFICATIONS (CTS)

DISCUSSION OF CHANGES (DOCs) TO THE CTS

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC) FOR LESS RESTRICTIVE CHANGES

MARKUP OF NUREG-1433, REVISION 1, SPECIFICATION

JUSTIFICATION FOR DIFFERENCES (JFDs) FROM NUREG-1433, REVISION 1

RETYPED PROPOSED IMPROVED TECHNICAL SPECIFICATIONS (ITS)

# JAFNPP

## IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

## ITS: 5.6

### Reporting Requirements

## MARKUP OF CURRENT TECHNICAL SPECIFICATIONS (CTS)

Section 5.6

A1



(C) Temporary changes to the procedures required by Specification 6.8.(A) may be made provided:

1. the intent of the original procedure is not altered.

2. the change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's license.

3. the change is documented, reviewed and approved by the appropriate member of plant management as required by Specification 6.5.0 within 14 days of implementation.

see ITS: Section 5.4

[5.6] 6.9 REPORTING REQUIREMENTS

(A) ROUTINE REPORTS

The following reports shall be submitted in accordance with 10 CFR 50.4 ~~50 unless otherwise noted.~~ A2

1. STARTUP REPORT

a. A summary report of plant startup and power escalation testing shall be submitted following (1) amendment to the license involving a planned increase in power level, (2) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (3) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The Startup Report for the initial fuel cycle shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report. Startup Reports for subsequent fuel cycles shall address startup tests that are necessary to demonstrate the acceptability of changes and modifications.

LA1

CTS Amend 254

A1

(A) ROUTINE REPORTS (Continued)

1. STARTUP REPORT (Continued)

b. ~~Startup Reports shall be submitted within (1) 90 days following completion of the startup test program, or (2) 90 days following resumption or commencement of commercial power operation, or whichever is earliest. If the Start-up Report does not cover both events, i.e., completion of startup test program and resumption or commencement of commercial power operation, supplementary reports shall be submitted at least every three months until both events are completed.~~

LA1

2. ANNUAL REPORTS

[5.6.1] a. ~~Annual~~ Radiation Occupational Exposure ~~Tabulation~~ Report

INSERT 254b-1

A tabulation on an annual basis of the number of station, utility and other personnel (including contractors) ~~receiving exposures greater than 100 mrem/yr and their associated man rem exposure~~ according to work and job functions, 1/ e.g., reactor operations and surveillance, in-service inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totaling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources ~~shall~~ should be assigned to specific major work functions.

A3

INSERT 254b-2

INSERT 254b-3

L1

INSERT 254b-4

~~b. Annual Report of S/RV Failures and Challenges~~ M3

~~An annual report of safety/relief valve failures and challenges will be submitted prior to March 1 of each year.~~

M1

A3

[5.6.4] 3. MONTHLY OPERATING REPORT

Routine ~~A~~ report ~~providing a narrative summary of facility operating experience, major safety-related maintenance, and other pertinent information,~~ ~~should~~ shall be submitted no later than the 15th of each month following the calendar month covered ~~to the USNRC Director, Office of Management Information and Program Control.~~

Statistics and shutdown

M2

~~1/ This tabulation supplements the requirements of § 20.407 of 10 CFR Part 20.~~ 10 CFR 20.2206

Amendment No. ~~32~~, ~~120~~ 130



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CTS Insert 254b-1 A3

for whom monitoring was performed, receiving an annual deep dose equivalent > 100 mrems and the associated collective deep dose equivalent (reported in person-rem)

CTS Insert 254b-2 A3

ionization chamber, thermoluminescent dosimeter (TLD), electronic dosimeter,

CTS Insert 254b-3 A3

deep dose equivalent

CTS Insert 254b-4 M1

The report, covering the previous calendar year, shall be submitted by April 30 of each year.

CTS Amend 254

A1

LA2

7. The Radioactive Effluent Release Report shall contain the cause for unavailability of any environmental sample required by Table 6.1.1 and shall identify the locations for obtaining replacement samples. This shall also include a revised figure(s) and table for the ODCM reflecting the new location(s). Refer to Specification 6.1.c.

8. The Radioactive Effluent Release Report shall contain new locations identified in the land use census in accordance with Specifications 6.2.b or 6.2.c.

9. The Radioactive Effluent Release Report shall contain the events leading to the condition which resulted in exceeding 10 curies for tanks specified in the Limiting Conditions for Operation, Section 2.5.a.

[5.6.2] d. Annual Radiological Environmental Operating Report

~~Routine~~ The Annual Radiological Environmental Operating Reports covering the operation of the unit during the previous calendar year shall be submitted ~~prior to May 1~~ by May 15 of each year. L2

The Annual Radiological Environmental Operating Reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period. The report shall include a comparison with preoperational studies, operational controls (as appropriate), and environmental surveillance reports from the previous five years, and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of the Land Use Census required by Specification 6.2 A4

The material provided shall be consistent with objectives outlined in ODCM and in 10CFR 50, App I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Reports shall include the results of analysis of all radiological environmental samples and of all measurements taken during the period ~~pursuant to Table 6.1.1~~, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion in the report, the report shall note and explain the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

at the locations specified in the table and figures in the ODCM

The reports shall also include the following: A summary description of the Radiological Environmental Monitoring Program; at least two legible maps* covering all sampling locations and keyed to a table giving distances and directions from the centerline of one reactor; the results of participation in the Interlaboratory Comparison Program required by Specification 6.3 (or appropriate EPA cross-check program code), and discussion of all analyses in which the LLD's required by Table 6.1-3 were not routinely achievable.

* One map shall cover stations near the site boundary; a second shall include the more distant stations.

LA2

Section 5.6 A1



7.0 ADMINISTRATIVE CONTROLS

See ITS Section 5.4

7.1 RESPONSIBILITY

a. The Site Executive Officer shall have direct responsibility for assuring the operation of the James A. FitzPatrick Plant is conducted in such a manner as to provide continuing protection to the environment and shall delegate in writing the succession to this responsibility during his absence.

b. Implementation of the Radiological Effluent Technical Specifications is the responsibility of the General Manager - Operations, with the assistance of the plant staff organization.

7.2 PROCEDURES

Written procedures and administrative policies shall be established, implemented and maintained that meet or exceed the requirements and recommendations of Section 5 "Facility Administrative Policies and Procedures" of ANSI 18.7-1972 and Regulatory Guide 1.33, November 1972, Appendix A. In addition, procedures shall be established, implemented and maintained for the PCP, ODCM, and Quality Control Program for effluent and environmental monitoring using the guidance in Regulatory Guide 4.1, Revision 1.

7.3 REPORTING REQUIREMENTS

A3

a. Planned Liquid and Gaseous Releases

The limits for radioactive materials contained in liquid and gaseous effluents are contained in Specifications 2.3, 3.3 and 3.4.

b. Environmental Samples Exceeding Limits of Table 6.1-2

When the limits of Table 6.1-2 are exceeded, refer to Specification 6.1.b for reporting requirements.

c. ~~Semiannual~~ Radioactive Effluent Release Report year

Prior to May 1 of each year in accordance with 10 CFR 50.36a

[5.6.3] ~~Routine~~ Radioactive Effluent Release Reports covering the operation of the unit during the previous ~~6 months of operation~~ shall be submitted ~~within 60 days after January 1 and July 1 of each year. The period of the first report shall begin with the date of initial criticality.~~ A3 L3

1. The Radioactive Effluent Release Report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the ~~unit~~ ~~using as guidance Regulatory Guide 1.21, Revision 1, June 1974, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants", with data summarized on a quarterly basis following the format of Appendix B thereof.~~ A4

Plant. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1

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[RETS]

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REVISION H

A1

2. The Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year may include an annual summary of meteorological data collected over the previous year. If the meteorological data is not included, ENO shall retain it on file and provide it to the U.S. Nuclear Regulatory Commission upon request. This same report shall include an assessment of the radiation doses* due to the radioactive liquid and gaseous effluents released from the unit or station during the previous clandar year to the public. All assumptions used in making these assessments (i.e., specific activity, exposure time and location) shall be included in these reports. The assessment of radiation doses shall be performed in accordance with the ODCM.

LA2

3. The Radioactive Effluent Release Reports shall include any change to the PCP or the ODCM made during the reporting period, as well as a listing of new locations for dose calculations and/or environmental monitoring indentified by the land use census pursuant to Specification 6.2.

4. The Radioactive Effluent Release Report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses* to the likely most exposed member of the public from reactor releases and other nearby uranium fuel cycle sources (including doses from primary effluent pathways and direct radiation) during the previous calendar year, to show conformance with 40 CFR 190, Environmental Radiation Protection Standards for Nuclear Power Operation. This assessment of radiation doses is performed in accordance with the ODCM.

5. The Radioactive Effluent Release Reports shall include the following information for each class of solid waste (defined by 10 CFR 61) shipped offsite during the report period:

(a) Container volume;

(b) Total curie quantity (specify whether determined by measurement or estimate),

(c) Principal radionuclides (specify whether determined by measurement or estimate),

(d) Source of waste and processing employed (e.g., dewatered spent resin, compacted dry waste, evaporator bottoms),

(e) Type of container (e.g., LSA, Type A, Large Quantity), and

(f) Solidification agent or absorbent (e.g., cement, Dow media, etc.)

6. The Radioactive Effluent Release Reports shall include a list and description of unplanned releases, to unrestricted areas of radioactive materials in gaseous and liquid effluents made during the reporting period.

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* The dose assessment sections of the Semiannual Radiological Effluent Release Report shall be submitted within 90 days after January 1 of each year as an addendum to the Semiannual Radiological Effluent Release Report.

Section 5.6 A1



(A) ROUTINE REPORTS (Continued)

4. CORE OPERATING LIMITS REPORT

[5.6.5.a] a. Core operating limits shall be established prior to startup from each reload cycle, or prior to any remaining portion of a reload cycle for the following:

A7

- The Average Planar Linear Heat Generation Rates (APLHGR) of Specification ~~3.5.H~~ 3.2.1
- The Minimum Critical Power Ratio (MCPR) ~~and MCPR low flow adjustment factor, $K_f$~~ of Specifications ~~3.1.B and 4.1.E~~ 3.2.2
- The Linear Heat Generation Rate (LHGR) of Specification ~~3.5.I~~ 3.2.3
- The Reactor Protection System (RPS) APRM Neutron Flux-High ~~flow biased~~ ~~trip settings~~ Allowable Value of Table ~~3.1-1~~ 3.3.1.1-1;
- The ~~flow biased APRM and~~ Rod Block Monitor (RBM) Upscale rod block ~~settings~~ of Table ~~3.2-3~~ 3.3.2.1-1 and

A8

- The Power/Flow Exclusion Region of Specification ~~3.5.J~~ 3.4.1

and shall be documented in the Core Operating Limits Report (COLR).

[5.6.5.b] b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC as described in:

A11

1. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A~~, 14, June 2000~~.
2. "James A. FitzPatrick Nuclear Power Plant SAFER/GESTR - LOCA Loss-of-Coolant Accident Analysis," NEDC-31317P~~, Revision 2, April 1993~~.
3. "BWR Owners' Group Long-term Stability Solutions Licensing Methodology," NEDO-31960-A~~, June 1991~~.
4. ~~"BWR Owners' Group Long-term Stability Solutions Licensing Methodology," NEDO-31960-A, Supplement 1, March 1992.~~

The COLR will contain the complete identification of each of the TS referenced topical reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements).

TSTF-363, R0

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ITS

[5.6.5.c] c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

[5.6.5.d] d. The COLR, including any mid-cycle revisions or supplements thereto, shall be provided, upon issuance for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

A10

RAI 5.6-1



eTS Amend 254

Section 5.6

A1



(B) SPECIAL REPORTS

A6

1. ~~Fifteen copies of the Evaluation Report of the results of the first five years of performance of the non-destructive inspection listed in Table 4.6-1 of Technical Specifications 4.6.F, Structural Integrity, relating to the FitzPatrick in-service inspection program shall be submitted to the NRC, Director of Operating Reactors, within three months of the completion of the fifth year of the program.~~

2. DELETED

See CTS Chapter 6.0

6.10 RECORD RETENTION

(A) The following records shall be retained for at least five years:

1. Records and logs of facility operation covering time intervals at each power level.

2. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

3. All Reportable Events.

4. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

5. Records of reactor tests and experiments.

6. Records of changes made to Operating Procedures.

7. Records of radioactive shipments.

8. Records of sealed source leak tests and results.

9. Records of annual physical inventory of all source material of record.

A9 - Add proposed Section 5.6.6, PAM Report

CTS Amend 254

# JAFNPP

## IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

## ITS: 5.6

Reporting Requirements

## DISCUSSION OF CHANGES (DOCs) TO THE CTS

ADMINISTRATIVE CHANGES

A1 In the conversion of the James A. FitzPatrick Nuclear Power Plant (JAFNPP) Current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes. Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the conventions in NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4", Revision 1 (i.e., Improved Standard Technical Specifications (ISTS)).

A2 CTS 6.9.A, "Routine Reports," requires that, "The following reports shall be submitted in accordance with 10 CFR 50 unless otherwise noted." ITS 5.6 requires that, "The following reports shall be submitted in accordance with 10 CFR 50.4." This change provides a more specific citation of the regulation involved. Since this change does not modify any technical requirements, it is administrative and has no adverse impact on safety.

A3 CTS 6.9.A.2 and CTS RETS 7.3.c are revised in ITS 5.6.1 and ITS 5.6.3 based on the letter from NRC (C. I. Grimes) to the four Owners' Groups, dated July 28, 1995, which proposed changes to the standard Technical Specifications to reflect the revised 10 CFR 20 and 10 CFR 50.36a requirements. The changes are also consistent with TSTF-152, Revision 0. Since this change involves revised reporting requirements and recognition of additional types of personnel monitoring instruments, it is therefore administrative and has no adverse impact on safety. TSTF-152, R0

A4 CTS RETS 7.3.c and 7.3.d provide a narrative description of the material required to be included in the Radioactive Effluent Release Report and the Annual Radiological Environmental Operating Report, respectively. ITS 5.6.2 and 5.6.3 require that the material provided in the reports be consistent with the objectives of the ODCM, the PCP, and with 10 CFR 50.36a and 10 CFR 50, Appendix I. In making this change, many of the details provided in the CTS are not retained in the ITS, since they are already covered within the cited programs and regulations. Since this change does not modify any technical requirements, it is administrative and has no adverse impact on safety.

A5 CTS RETS 7.3.a and 7.3.b are not retained in the ITS. These Specifications only refer to other specifications and need not be repeated in the ITS. Since this change does not modify any technical requirements, it is administrative and has no adverse impact on safety.

ADMINISTRATIVE CHANGES

A6 CTS 6.9.B.1 requires that a report of the results of the first five years of performance of the non-destructive inspections listed in CTS Table 4.6-1 of CTS 4.6.F, Structural Integrity, be submitted to the NRC within three months of the completion of the fifth year of the program. This Specification is not retained in the ITS. The inspection activities have been performed with satisfactory results, and the report submitted to the NRC. Since this change involves deletion of requirements that have been fulfilled, it is administrative and has no adverse impact on safety.

A7 CTS 6.9.A.4.a reference to the MCPR low flow adjustment factor $K_f$ has been deleted. It is not necessary to identify this particular function since the process of determining the MCPR in the COLR includes the requirement to multiply the MCPR by the appropriate $K_f$. Therefore, this change is administrative.

A8 CTS 6.9.A.4.a reference to flow biased APRM rod blocks has been deleted since this function has been relocated. The relocation of the CTS Table 3.2-3 APRM flow biased function is discussed in ITS 3.3.2.1 Discussion of Changes (R1). Therefore, this change is administrative.

A9 ITS 5.6.6 has been added which requires a special report to be submitted for Post Accident Monitoring Instrumentation when Required Actions and associated Completion Times cannot be met or more than one channel is inoperable for specific instrumentation. The report is referenced in Actions B and F of ITS 3.3.3.1, Post Accident Monitoring Instrumentation. Changes to the current requirements of the Post Accident Monitoring Instrumentation Specification are described in the Discussion of Changes for ITS 3.3.3.1, "PAM Instrumentation". Since the addition of the reporting requirement is a direct result of an added Action, this change is considered administrative.

RAI 5.6-1

A10 CTS 6.9.A.4.d details concerning distribution of the COLR are being deleted since the details are a duplication of the document distribution requirements of 10 CFR 50.4. This change does not change any technical requirements and it is therefore administrative and does not have an adverse impact on safety.

A11 CTS 6.9.A.4.b details regarding the revision number and date of the topical reports used to determine the core operating limits are being deleted since the complete identification of the topical reports used to prepare the COLR are to be identified in the COLR. The changes are also consistent with TSTF-363, Revision 0 and a letter from S. A. Richards (NRC) to J. F. Malley (Siemens Power Corporation) dated December 15, 1999.

TSTF-363,R0

## TECHNICAL CHANGES - MORE RESTRICTIVE

M1 CTS 6.9.A.2.a requires submittal of an Annual Occupational Exposure Tabulation on an annual basis. ITS 5.6.1 requires that an Occupational Radiation Exposure Report be submitted by April 30 of each year. This change adopts a specific date for submittal of a report, which is more restrictive. Since this change involves submittal of an after-the-fact report, and does not affect plant operation, it has no adverse impact on safety.

M2 CTS 6.9.A.3 requires that the Monthly Operating Report should be submitted by the 15th of each month. ITS 5.6.4 requires that the Monthly Operating Report shall be submitted by the 15th of the month. This change revises an action from "should" to "shall," and is therefore more restrictive. Since this change involves submittal of an after-the-fact report, and does not affect plant operation, it has no adverse impact on safety.

M3 CTS 6.9.A.2.b requirement, to submit an annual report for all failures and challenges to safety/relief valves, has been moved to proposed ITS 5.6.4 for monthly reports. The report is required on a monthly basis in ITS instead of the current annual basis. Therefore, this change is a more restrictive change necessary to achieve consistency with NUREG-1433, Revision 1. This change has no adverse impact on safety.

## TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA1 The details associated with CTS 6.9.A.1, "STARTUP REPORT," are proposed to be relocated to the UFSAR. The Startup Report is a summary of plant startup and power escalation testing following receipt of the operating license, increase in licensed power level, installation of nuclear fuel with a different design or manufacturer than the current fuel, and modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The report provides the NRC a mechanism to review the appropriateness of licensee activities after-the-fact, but provides no regulatory authority once the report is submitted (i.e., no requirement for NRC approval). The Quality Assurance requirements of 10 CFR 50, Appendix B and the Startup Test Program provisions contained in the UFSAR provide assurance that the listed activities will be adequately performed and that appropriate corrective actions, if required, are taken. Given that the report was required to be provided to the Commission no later than 90 days following completion of the respective milestone, report completion and submittal was clearly not necessary to assure operation of the facility in a safe manner for the interval between completion of the startup testing and submittal of the report. Additionally, given there is no

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 CTS 6.9.A.2.a requires submittal of an Annual Occupational Exposure Tabulation on an annual basis. ITS 5.6.1 requires that an Occupational Radiation Exposure Report be submitted by April 30 of each year. This change adopts a specific date for submittal of a report, which is more restrictive. Since this change involves submittal of an after-the-fact report, and does not affect plant operation, it has no adverse impact on safety.

M2 CTS 6.9.A.3 requires that the Monthly Operating Report should be submitted by the 15th of each month. ITS 5.6.4 requires that the Monthly Operating Report shall be submitted by the 15th of the month. This change revises an action from "should" to "shall," and is therefore more restrictive. Since this change involves submittal of an after-the-fact report, and does not affect plant operation, it has no adverse impact on safety.

M3 CTS 6.9.A.2.b requirement, to submit an annual report for all failures and challenges to safety/relief valves, has been moved to proposed ITS 5.6.4 for monthly reports. The report is required on a monthly basis in ITS instead of the current annual basis. Therefore, this change is a more restrictive change necessary to achieve consistency with NUREG-1433, Revision 1. This change has no adverse impact on safety.

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA1 The details associated with CTS 6.9.A.1, "STARTUP REPORT," are proposed to be relocated to the UFSAR. The Startup Report is a summary of plant startup and power escalation testing following receipt of the operating license, increase in licensed power level, installation of nuclear fuel with a different design or manufacturer than the current fuel, and modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The report provides the NRC a mechanism to review the appropriateness of licensee activities after-the-fact, but provides no regulatory authority once the report is submitted (i.e., no requirement for NRC approval). The Quality Assurance requirements of 10 CFR 50, Appendix B and the Startup Test Program provisions contained in the UFSAR provide assurance that the listed activities will be adequately performed and that appropriate corrective actions, if required, are taken. Given that the report was required to be provided to the Commission no later than 90 days following completion of the respective milestone, report completion and submittal was clearly not necessary to assure operation of the facility in a safe manner for the interval between completion of the startup testing and submittal of the report. Additionally, given there is no

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA1 (continued)

requirement for the Commission to approve the report, the Startup Report is not necessary to assure operation of the facility in a safe manner. Therefore, the relocated details are not required to be in the ITS to provide adequate protection of the public health and safety. Based on these considerations, the Startup Report may be removed from Technical Specifications and relocated to the UFSAR. Changes to the UFSAR are controlled by the provisions of 10 CFR 50.59.

LA2 The details in CTS RETS 7.3.c and 7.3.d, associated with the contents of the Radioactive Effluent Release Report (ITS 5.6.3), and the Annual Radiological Environmental Operating Report (ITS 5.6.2), are being relocated to the Offsite Dose Calculation Manual (ODCM). This change is based on Generic Letter 89-01 which provides guidance on the removal of the Radiological Environmental Technical Specifications from the Technical Specifications. Therefore, the relocated details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the ODCM will be controlled in accordance with ITS 5.5.1, Offsite Dose Calculation Manual (ODCM).

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CTS 6.9.A.2.a requires that, "at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions." ITS 5.6.1 requires that, "at least 80% of the total whole body dose received from external sources should be assigned to specific major work functions." This is a relaxation of requirements, and is less restrictive. This change is acceptable, however, because this requirement relates to the format and content of an annual occupational radiation exposure report, and has no impact on the operation of the plant. This change is consistent with NUREG-1433, Revision 1.

L2 CTS RETS 7.3.d requires that the Annual Radiological Environmental Operating Report be submitted prior to May 1 of each year. ITS 5.6.2 requires this report to be submitted by May 15 of each year. This is a relaxation of requirements, which is less restrictive. This change is acceptable, however, because the report is after-the-fact, covering the previous calendar year, and there is no requirement for the NRC to approve the report. Completion and submittal of the reports is clearly not necessary to ensure safe operation of the plant during the additional time intervals provided by these changes. This change is consistent with NUREG-1433, Revision 1.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L3 CTS RETS 7.3.c requires that the Radioactive Effluent Release Report be submitted on a semiannual basis. ITS 5.6.3 requires that this report be submitted in accordance with 10 CFR 50.36a, on an annual basis. This is a relaxation of requirements, which is less restrictive. This change is acceptable, however, since this report covers the previous calendar year, and there is no requirement for the NRC to approve the report. Completion and submittal of the report is clearly not necessary to ensure safe operation of the plant during the additional time interval provided by the proposed change. This change is consistent with NUREG-1433, Revision 1.

TECHNICAL CHANGES - RELOCATIONS

None

# JAFNPP

## IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

## ITS: 5.6

Reporting Requirements

## NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC) FOR LESS RESTRICTIVE CHANGES

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 5.6 - REPORTING REQUIREMENTS

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CHANGE

New York Power Authority has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion 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?

   The proposed change does not involve any physical alteration of plant systems, structures, or components, changes in parameters governing normal plant operation, or methods of operation. The proposed change revises a reporting requirement from "shall" to "should." Reporting requirements are not considered to be initiators of accidents, nor do the affect plant operations. Therefore, the proposed change does not involve an increase in the probability or consequence of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant systems, structures, or components, changes in parameters governing normal plant operation, or methods of operation. The proposed change does not introduce any new modes of operation. A reporting requirement cannot be the initiator of a new or different kind of accident. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The proposed change does not introduce any new modes of operation. There are no margins of safety related to any safety analyses that are dependent upon the proposed change. Therefore, this change does not involve a reduction in a margin of safety.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L2 CHANGE

New York Power Authority has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion 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?

   The proposed change does not involve any physical alteration of plant systems, structures, or components, changes in parameters governing normal plant operation, or methods of operation. The proposed change modifies the dates for submittal of "after the fact" information, which has no impact on the safety analysis. Since these reports cover the previous calendar year, and there is no requirement for the NRC to approve these reports, completion and submittal of the reports is clearly not necessary to ensure safe operation of the plant during the additional time intervals provided by the proposed change. Therefore, the proposed change does not involve an increase in the probability or consequence of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant systems, structures, or components, changes in parameters governing normal plant operation, or methods of operation. The proposed change does not introduce any new modes of operation. The proposed change impacts only the administrative requirements for submittal of information and does not directly impact operation of the plant. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The proposed change does not introduce any new modes of operation. There are no margins of safety related to any safety analyses that are dependent upon the proposed change. Therefore, this change does not involve a reduction in a margin of safety.

## TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

### L3 CHANGE

New York Power Authority has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion 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?

   The proposed change does not involve any physical alteration of plant systems, structures, or components, changes in parameters governing normal plant operation, or methods of operation. The proposed change modifies the date for submittal of "after the fact" information, which clearly has no impact on the safety analysis. Since this report covers the previous calendar year, and there is no requirement for the NRC to approve the report, completion and submittal of the report is not necessary to ensure safe operation of the plant during the additional time interval provided by the proposed change. Therefore, the proposed change does not involve an increase in the probability or consequence of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant systems, structures, or components, changes in parameters governing normal plant operation, or methods of operation. The proposed change does not introduce any new modes of operation. The proposed change impacts only the administrative requirements for submittal of information and does not directly impact operation of the plant. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The proposed change does not introduce any new modes of operation. There are no margins of safety related to any safety analyses that are dependent upon the proposed change. Therefore, this change does not involve a reduction in a margin of safety.

# JAFNPP

## IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

## ITS: 5.6

Reporting Requirements

MARKUP OF NUREG-1433, REVISION 1
SPECIFICATION

## 5.0 ADMINISTRATIVE CONTROLS

CTS

### 5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

[6.9.A.2] 5.6.1 Occupational Radiation Exposure Report

~~NOTE~~
~~A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.~~

PA1

TA2 X1

INSERT 5.6.1-1

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures > 100 mrem/yr and their associated man rem exposure according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance [describe maintenance], waste processing, and refueling). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket dosimeter, thermoluminescent dosimeter (TLD), or film badge measurements. Small exposures totalling < 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources should be assigned to specific major work functions. The report shall be submitted by April 30 of each year. [The initial report shall be submitted by April 30 of the year following initial criticality.]

[RETS 3.7.d] 5.6.2 Annual Radiological Environmental Operating Report

~~NOTE~~
~~A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.~~

PA1

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual

(continued)



TYP all pages

INSERT 5.6.1-1

TA2

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors), for whom monitoring was performed, receiving an annual deep dose equivalent >100 mrems and the associated collective deep dose equivalent (reported in person - rem) according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket ionization chamber, thermoluminescence dosimeter (TLD), electronic dosimeter, or film badge measurements. Small exposures totaling <20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total deep dose equivalent received from external sources should be assigned to specific major work functions. The report covering the previous calendar year shall be submitted by April 30 of each year. ~~[The initial report shall be submitted by April 30 of the year following the initial criticality.]~~

PA3

PA3

PA2

TSTF-156, R0

CTS

## 5.6 Reporting Requirements

[RETS 7.3.d] 5.6.2 Annual Radiological Environmental Operating Report (continued)

(ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period at ~~pursuant to~~ the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements (in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979). ~~[The report shall identify the TLD results that represent collocated dosimeters in relation to the NRC TLD program and the exposure period associated with each result.]~~ In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

PA3 CLB1 TA4 TSTF-348, R0

5.6.3 Radioactive Effluent Release Report

[RETS 7.3.c]

~~NOTE: A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.~~ PA1

The Radioactive Effluent Release Report covering the operation of the ~~unit~~ plant during the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the ~~unit~~ plant. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR part 50, Appendix I, Section IV.B.1.

TA2 X1 PA1 TA2 X1 PA1 TA2 TSTF-152, R0

[6.9.A.3] 5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience~~s, including documentation of all challenges to the safety/relief~~

TA3 TSTF-258, R4

(continued)

CTS

## 5.6 Reporting Requirements

[6.9.A.3] 5.6.4 Monthly Operating Reports (continued)

TA3

TSTF-258, R4

valves, shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

[6.9.A.4] 5.6.5 CORE OPERATING LIMITS REPORT (COLR)

[6.9.A.4.a]

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

INSERT 5.6.5-1

~~[The individual specifications that address core operating limits must be referenced here.]~~

[6.9.A.4.b]

CLB3

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

INSERT 5.6.5-2

TSTF-363, R0

~~[Identify the Topical Report(s) by number, title, date, and NRC staff approval document, or identify the staff Safety Evaluation Report for a plant specific methodology by NRC letter and date.]~~

TA5

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

[A10]

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

X2

~~5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)~~

~~a. RCS pressure and temperature limits for heatup, cooldown, low temperature operation, critically, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:~~

~~(continued)~~

ITS Insert 5.6.5-1 CLB3

1. The AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGR) of Specification 3.2.1;

2. The MINIMUM CRITICAL POWER RATIO (MCPR) of Specification 3.2.2;

3. The LINEAR HEAT GENERATION RATE (LHGR) of Specification 3.2.3;

4. The Reactor Protection System (RPS) APRM Neutron Flux-High (Flow Biased) Allowable Value of Table 3.3.1.1-1;

5. The Rod Block Monitor-Upscale Allowable Value of Table 3.3.2.1-1; and

6. The Power/Flow Exclusion Region of Specification 3.4.1.

ITS Insert 5.6.5-2 TA5

TSTF-363, R0

1. NEDE-24011-P-A, General Electric Standard Application for Reactor Fuel;

2. NEDC-31317P, James A. FitzPatrick Nuclear Power Plant SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis; and

3. NEDO-31960-A, BWR Owners' Group Long-Term Stability Solutions Licensing Methodology.

The COLR will contain the complete indentification for each of the TS referenced topical reports used to prepare the COLR (i.e., report number, title, revision, date and any supplements).

## 5.6 Reporting Requirements

SX3

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued)

[The individual specifications that address RCS pressure and temperature limits must be referenced here.]

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents: [Identify the NRC staff approval document by date.]

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

Reviewers' Notes: The methodology for the calculation of the P-T limits for NRC approval should include the following provisions:

1. The methodology shall describe how the neutron fluence is calculated (reference new Regulatory Guide when issued).
2. The Reactor Vessel Material Surveillance Program shall comply with Appendix H to 10 CFR 50. The reactor vessel material irradiation surveillance specimen removal schedule shall be provided, along with how the specimen examinations shall be used to update the PTLR curves.
3. Low Temperature Overpressure Protection (LTOP) System lift setting limits for the Power Operated Relief Valves (PORVs), developed using NRC-approved methodologies may be included in the PTLR.
4. The adjusted reference temperature (ART) for each reactor beltline material shall be calculated, accounting for radiation embrittlement, in accordance with Regulatory Guide 1.99, Revision 2.
5. The limiting ART shall be incorporated into the calculation of the pressure and temperature limit curves in accordance with NUREG-0800 Standard Review Plant 5.3.2, Pressure-Temperature Limits.
6. The minimum temperature requirements of Appendix G to 10 CFR Part 50 shall be incorporated into the pressure and temperature limit curves.

(continued)

CTS

## 5.6 Reporting Requirements

(X2)

5.6.6 ~~Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued)~~

~~7. Licensees who have removed two or more capsules should compare for each surveillance material the measured increase in reference temperature ($RT_{NDT}$) to the predicted increase in $RT_{NDT}$; where the predicted increase in $RT_{NDT}$ is based on the mean shift in $RT_{NDT}$ plus the two standard deviation value ($2\sigma_{\Delta}$) specified in Regulatory Guide 1.99, Revision 2. If the measured value exceeds the predicted value (increase in $RT_{NDT}$ + $2\sigma_{\Delta}$), the licensee should provide a supplement to the PTLR to demonstrate how the results affect the approved methodology.~~

CLB2

TA1

5.6.7 ~~EDG Failures Report~~

~~If an individual emergency diesel generator (EDG) experiences four or more valid failures in the last 25 demands, these failures and any nonvalid failures experienced by that EDG in that time period shall be reported within 30 days. Reports on EDG failures shall include the information recommended in Regulatory Guide 1.9, Revision 3, Regulatory Position C.5, or existing Regulatory Guide 1.108 reporting requirement.~~

CLB2

TA1

5.6.8 6 PAM Report

When a report is required by Condition B or G F of LCO 3.3.3.1, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

PA4

TSTF-37, R2

X3

~~Reviewer's Note: These reports may be required covering inspection, test, and maintenance activities. These reports are determined on an individual basis for each unit and their preparation and submittal are designated in the Technical Specifications.~~

# JAFNPP

## IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

## ITS: 5.6

Reporting Requirements

JUSTIFICATION FOR DIFFERENCES (JFDs) FROM NUREG-1433, REVISION 1

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS: 5.6 - REPORTING REQUIREMENTS

Editorial TSTF-258, R4

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 ITS 5.6.2 bracketed statement, related to collocated dosimeters and the NRC TLD program, is deleted consistent with current licensing basis of CTS RETS 7.3.d. This is also consistent with TSTF-348, Revision 0.

TSTF-37, R0

CLB2 ISTS Specification 5.6.7, "EDG Failures Report," is not adopted, consistent with current licensing basis. Subsequent Specifications are renumbered accordingly. This change is also consistent with TSTF-37, Revision 2.

TSTF-363, R0

CLB3 ITS 5.6.5.a has been revised to reflect the current licensing requirements of the JAFNPP core operating limits report (COLR) consistent with CTS 6.9.A.4.a.

PLANT SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

PA1 The ITS 5.6.1, 5.6.2, and 5.6.3 bracketed NOTES referring to multiple unit stations have been deleted since JAFNPP is a single unit plant site.

PA2 The ITS 5.6.1 statement, referring to submittal of first report following initial criticality, is deleted since JAFNPP has been licensed since 1974.

PA3 Editorial changes have been made for enhanced clarity or to correct a grammatical/typographical error: the phrase, "pursuant to," is replaced with the term, "at."

PA4 Changes have been made to reflect proper Condition or LCO.

PA5 Wording preference: the term, "unit," is replaced with the term, "plant."

PLANT SPECIFIC DIFFERENCE IN DESIGN OR DESIGN BASIS (DB)

None

TSTF-37, R2

DIFFERENCE BASED ON APPROVED TRAVELER (TA)

TA1 The changes presented in TSTF Technical Specification Change Traveler number 37, Revision 2, have been incorporated into the revised Improved Technical Specifications.

DIFFERENCE BASED ON APPROVED TRAVELER (TA) (continued)

TSTFs as stated

TA2 The changes presented in TSTF Technical Specification Change Traveler number 152, Revision 0, have been incorporated into the revised Improved Technical Specifications.

TA3 The changes presented in TSTF Technical Specification Change Traveler number 258, Revision 4, have been incorporated into the revised Improved Technical Specifications.

TA4 The changes presented in TSTF Technical Specification Change Traveler number 348, Revision 0, have been incorporated into the revised Improved Technical Specifications.

TA5 The changes presented in TSTF Technical Specification Change Traveler number 363, Revision 0, have been incorporated into the revised Improved Technical Specifications.

DIFFERENCE BASED ON PENDING TRAVELER (TP)

None

DIFFERENCE FOR ANY OTHER REASON THAN ABOVE (X)

TSTF-152, R0

X1 Changes to ITS Specifications 5.6.1 and 5.6.3 are made based on letter from NRC (C. I. Grimes) to the four Owners' Groups, dated July 28, 1995, which proposed changes to the standard Technical Specifications to reflect the revised 10 CFR 20 and 10 CFR 50.36a. The changes are also consistent with TSTF-152, Revision 0.

X2 ISTS 5.6.6 RCS Pressure and Temperature Limits Report is not included in the JAFNPP ITS since NRC approved methodology for the development of RCS pressure and temperature limits does not exist at JAFNPP. Subsequent Specifications have been renumbered accordingly.

X3 The bracketed "Reviewer's Note" has been deleted. This information is for the NRC reviewer to understand exactly what is needed to meet this requirement. This is not meant to be retained in the final version of the plant specific information.

# JAFNPP

## IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

## ITS: 5.6

Reporting Requirements

RETYPED PROPOSED IMPROVED TECHNICAL SPECIFICATIONS (ITS)

# 5.0 ADMINISTRATIVE CONTROLS

## 5.6 Reporting Requirements

TSTF-152, R0

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 Occupational Radiation Exposure Report

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors), for whom monitoring was performed, receiving an annual deep dose equivalent > 100 mrem and the associated collective deep dose equivalent (reported in person-rem) according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket ionization chamber, thermoluminescent dosimeter (TLD), electronic dosimeter, or film badge measurements. Small exposures totalling < 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total deep dose equivalent received from external sources should be assigned to specific major work functions. The report, covering the previous calendar year shall be submitted by April 30 of each year.

5.6.2 Annual Radiological Environmental Operating Report

The Annual Radiological Environmental Operating Report covering the operation of the plant during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period at the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be

(continued)

## 5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued)

submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 Radioactive Effluent Release Report

The Radioactive Effluent Release Report covering the operation of the plant during the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the plant. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1. TSTF-152, R0

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report. TSTF-258, R4

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. The AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGR) of Specification 3.2.1;

2. The MINIMUM CRITICAL POWER RATIO (MCPR) of Specification 3.2.2;

3. The LINEAR HEAT GENERATION RATE (LHGR) of Specification 3.2.3;

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

4. The Reactor Protection System (RPS) APRM Neutron Flux–High (Flow Biased) Allowable Value of Table 3.3.1.1-1;

5. The Rod Block Monitor-Upscale Allowable Value of Table 3.3.2.1-1; and

6. The Power/Flow Exclusion Region of Specification 3.4.1.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following document:

1. NEDE-24011-P-A, General Electric Standard Application for Reactor Fuel;

2. NEDC-31317P, James A. FitzPatrick Nuclear Power Plant SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis; and

3. NEDO-31960-A, BWR Owners' Group Long-Term Stability Solutions Licensing Methodology.

TSTF-363, R0

The COLR will contain the complete identification for each of the Technical Specification referenced topical reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements).

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

(continued)

## 5.6 Reporting Requirements (continued)

### 5.6.6 PAM Report

When a report is required by Condition B or F of LCO 3.3.3.1, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

# JAFNPP

## IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

## ITS: 5.7

High Radiation Area

MARKUP OF CURRENT TECHNICAL SPECIFICATIONS (CTS)

DISCUSSION OF CHANGES (DOCs) TO THE CTS

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC) FOR LESS RESTRICTIVE CHANGES

MARKUP OF NUREG-1433, REVISION 1, SPECIFICATION

JUSTIFICATION FOR DIFFERENCES (JFDs) FROM NUREG-1433, REVISION 1

RETYPED PROPOSED IMPROVED TECHNICAL SPECIFICATIONS (ITS)

# JAFNPP

## IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

## ITS: 5.7

High Radiation Area

## MARKUP OF CURRENT TECHNICAL SPECIFICATIONS (CTS)

Specification 5.7

A1

see CTS Chapter 6.0

(B) The following records shall be retained for the duration of the Facility Operating License:

1. Records of any drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.

2. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

3. Records of facility radiation and contamination surveys.

4. Records of radiation exposure for all individuals entering radiation control areas.

5. Records of gaseous and liquid radioactive material released to the environs.

6. Records of transient or operational cycles for those facility components identified in Table 6.10-1.

See ITS Section 5.5

7. Records of training and qualification for current members of the plant staff.

8. Records of in-service inspections performed pursuant to these Technical Specifications.

9. Records of Quality Assurance activities required by the Quality Assurance Manual.

10. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

11. Records of meetings of the PORC and the SRC.

12. Records for Environmental Qualification which are covered under the provisions of paragraph 6.15.

13. Records of the service life of all hydraulic and mechanical snubbers, whose failure could adversely affect any safety-related system, including the date at which the service life commences and associated installation and maintenance records as of the effective date of this ammendment.

see CTS Chapter 6.0

## 6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared and adhered to for all plant operations. These procedures shall be formulated to maintain radiation exposures received during operation and maintenance as far below the limits specified in 10 CFR 20 as practicable. The procedures shall include planning, preparation, and training for operation and maintenance activities. They shall also include exposure allocation, radiation and contamination control techniques, and final debriefing.

LA1

Amendment No. ~~22~~, ~~37~~, 92
Order dated October 24, 1980

Section 5.7

A.1



add proposed ITS Section 5.7

L1

[5.7] 6.11 (A) High Radiation Area

1. In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20, each High Radiation Area (i.e., ≥100 mrem/hr) in which the intensity of radiation is 1000 mrem/hr or less shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP).* Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

    a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

    b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

    c. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility health physicist in the Radiation Work Permit.

2. The requirements of 6.11.A.1 above, shall also apply to each high radiation area in which the intensity of radiation is greater than 1000 mrem/hr. In addition, locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of the Shift Manager on duty and/or the radiation protection manager.

*Radiation protection personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas.

CTS Amend 270

# JAFNPP

## IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

## ITS: 5.7

High Radiation Area

## DISCUSSION OF CHANGES (DOCs) TO THE CTS

## ADMINISTRATIVE CHANGES

A1 In the conversion of the James A. FitzPatrick Nuclear Power Plant (JAFNPP) Current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes. Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the conventions in NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4", Revision 1 (i.e., Improved Standard Technical Specifications (ISTS)).

## TECHNICAL CHANGES - MORE RESTRICTIVE

None

## TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA1 The details in CTS 6.11 concerning the requirements of the Radiation Protection Program are proposed to be relocated to the UFSAR. CTS 6.11 requires procedures to be prepared for personnel radiation protection to maintain radiation exposures received during operation and maintenance as far below the limits specified in 10 CFR 20 as practicable. The procedures shall include planning, preparation, and training for operation and maintenance activities. They shall also include exposure allocation, radiation and contamination control techniques, and final debriefing. These procedures are for nuclear plant personnel and have no impact on nuclear safety or the health and safety of the public. Requirements to have a Radiation Protection Program are contained in 10 CFR 20.1101(a). Requirements to have procedures and engineering controls based on sound radiation protection principles to achieve occupational doses and dose to the members of the public that are as low as is reasonably achievable are contained in 10 CFR 20.1101(b). Since the CTS requirements are governed by regulation, there is no need to repeat them in the ITS to provide adequate protection of public health and safety. Changes to the UFSAR will be controlled by the provisions of 10 CFR 50.59.

## TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CTS 6.11.A, which provides high radiation area access control alternatives pursuant to 10 CFR 20.203(c)(2) (revised 10 CFR 20.1601(c)), has been significantly revised as a result of the changes to 10 CFR 20, the guidance provided in Regulatory Guide 8.38 (Control of

TSTF-258, R4

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 (continued)

Access to High and Very High Radiation Areas in Nuclear Power Plants), and current industry technology in controlling high radiation areas. The changes incorporated in ITS 5.7 (which are consistent with TSTF-258, Revision 4) include:

a capping dose rate to differentiate a high radiation area from a very high radiation area,

additional requirements for groups entering high radiation areas, and

clarification of the need for control of workers in high radiation areas.

As a result, this change will not decrease the ability to provide control of exposures from external sources in restricted areas.

TECHNICAL CHANGES - RELOCATIONS

None

# JAFNPP

## IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

## ITS: 5.7

High Radiation Area

## NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC) FOR LESS RESTRICTIVE CHANGES

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 5.7 - HIGH RADIATION AREA

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CHANGE

New York Power Authority has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion 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?

   The proposed alternatives for control of access to high radiation areas are consistent with the intent of 10 CFR 20.1601(a) and (b). The proposed changes do not affect the probability of an accident. The controls used for access to high radiation areas are not assumed in the initiation of any analyzed event. Also, the consequences of an accident are not affected by these changes. These changes are both consistent with good radiological practice and will provide an adequate level of radiation protection. These proposed changes do not impact the assumptions of any design basis accident. These changes will not alter assumptions relative to the mitigation of an accident or transient event. These changes have no impact on safe operation of the plant. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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 create the possibility of an accident. This change will not physically alter the plant (no new or different type of equipment or system will be installed). The changes in methods governing normal plant operations are consistent with the current safety analysis assumptions. Therefore, this change 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?

   The proposed alternatives for control of access to high radiation areas are consistent with the intent of 10 CFR 20.1601(a) and (b). The margin of safety is not reduced due to these proposed changes. These changes are both consistent with good radiological safety practice and have been found to provide an adequate level of radiation protection. In

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CHANGE

3. (continued)

addition, these changes provide the benefit of ensuring radiation dose to all workers is minimized by providing the flexibility to select the best means of providing a barrier and access control to a high radiation area, given the plant location and radiological conditions. These proposed changes have no impact on the safe operation of the plant. No change in analytic limits or setpoints is introduced by this change. The safety analysis assumptions will still be maintained, thus no question of safety exists. Therefore, these changes do not involve a significant reduction in a margin of safety.

# JAFNPP

## IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

## ITS: 5.7

High Radiation Area

MARKUP OF NUREG-1433, REVISION 1
SPECIFICATION

TAZ

INSERT 5.7

5.0 ADMINISTRATIVE CONTROLS

{5.7 High Radiation Area}

5.7.1 Pursuant to 10 CFR 20, paragraph 20.1601(c), in lieu of the requirements of 10 CFR 20.1601, each high radiation area, as defined in 10 CFR 20, in which the intensity of radiation is > 100 mrem/hr but < 1000 mrem/hr, shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., [Health Physics Technicians]) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates ≤ 1000 mrem/hr, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas.

Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device that continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel are aware of them.

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the [Radiation Protection Manager] in the RWP.

5.7.2 In addition to the requirements of Specification 5.7.1, areas with radiation levels ≥ 1000 mrem/hr shall be provided with locked or continuously guarded doors to prevent unauthorized entry and the keys shall be maintained under the administrative control of the Shift Foreman on duty or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP that shall specify the dose rate levels in the immediate work

TSTF-258, R4

(continued)

## 5.7 High Radiation Area

5.7.2 (continued)

areas and the maximum allowable stay times for individuals in those areas. In lieu of the stay time specification of the RWP, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

5.7.3 For individual high radiation areas with radiation levels of > 1000 mrem/hr, accessible to personnel, that are located within large areas such as reactor containment, where no enclosure exists for purposes of locking, or that cannot be continuously guarded, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded and conspicuously posted, and a flashing light shall be activated as a warning device.

TSTF-258, R.4

TA2

INSERT 5.7

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20:

5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from Any Surface Penetrated by the Radiation:

PA1

entrance or access point

(e.g., roped off)

a. Each ~~entryway~~ to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

PA2

PA3

, or be continuously guarded to prevent unauthorized personnel entry

b. Access to, and activities in, each such area shall be controlled by means of a Radiation Work Permit (RWP) or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided ~~that~~ they are following plant radiation protection procedures for entry to, exit from, and work in such areas.

PA1

TSTF-258, R4

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously displays radiation dose rates in the area; or

2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the devices's dose alarm setpoint is reached, with a pre-set ~~an appropriate~~ alarm setpoint, or

PA2

3. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area, or

TA2

4. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

   (a) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

   (b) Be under the surveillance, as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.

PA2

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

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TA 2

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation:

PA 1

a. Each ~~entryway~~ entrance or access point to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked door or gate that prevents unauthorized entry, or be continuously guarded to prevent unauthorized personnel entry and, in addition:

PA 2 PA 3

1. All such door and gate keys shall be maintained under the administrative control of the shift supervisor, radiation protection manager, or his or her designee.

2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.

b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided ~~that~~ they are following plant radiation protection procedures for entry to, exit from, and work in such areas.

PA 1

d. Each individual or group entering such an area shall possess:

PA 2

1. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with ~~an appropriate~~ a pre-set alarm setpoint, or

2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area withindividual the means to communicate with and control every individual in the area, or

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TA 2

3. A self reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

    (a) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

    (b) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area, or,

4. In those cases where options 2 and 3, above, are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle, a radiation monitoring device that continuously displays radiation dose rates in the area may be used.

TSTF-258, R4

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive pre-job breifing knowledge, and pre-job briefing does not require documentation prior to initial entry. prior to entry into such areas. This dose rate determination, knowledge and pre-job briefing does not require documentation prior to entry.

f. Such individual areas that are within a larger area where no enclosure exists for the purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate, nor continuously guarded, but shall be barricaded, conspicuously posted, and a clearly visible flashing light shall be activated at the area as a warning device.

# JAFNPP

## IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

## ITS: 5.7

High Radiation Area

JUSTIFICATION FOR DIFFERENCES (JFDs)
FROM NUREG-1433, REVISION 1

RETENTION OF EXISTING REQUIREMENT (CLB)

None.

PLANT SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

editorial

PA1 ITS 5.7. High Radiation Area presentation format has been revised consistent with the Writer's Guide for the Restructrued Technical Specifications.

PA2 Editorial changes have been made for enhanced clarity or to correct a grammatical/typographical error.

PA3 ITS 5.7 (as modified by TSTF-258, R4) have been revised to allow continuous guarding of a high radiation area entrance or access point in lieu of the area being barricaded and conspicuously posted (ITS 5.7.1.a) or in lieu of the area being provided with a locked door or gate to provide positive control over personnel entry to the high radiation area (ITS 5.7.2.a). This allowance is necessary for numerous reasons, including when there is a transitory high radiation area, when there is a discovery of a new high radiation area, when establishing a temporary entrance or access point to a high radiation area, and when a door or gate is not functioning as intended (e.g., a door lock is broken).

PLANT SPECIFIC DIFFERENCE IN DESIGN OR DESIGN BASIS (DB)

None.

DIFFERENCE BASED ON APPROVED TRAVELER (TA)

TSTF-258, R4

TA1 Not used.

TA2 The changes presented in TSTF Techncial Specification Change Traveler number 258, Revision 4, have been incorporated into the revised Improved Technical Specifications.

DIFFERENCE BASED ON PENDING TRAVELER (TP)

None.

DIFFERENCE FOR OTHER REASONS THAN ABOVE (X)

X1 Not Used.

# JAFNPP

## IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

## ITS: 5.7

High Radiation Area

RETYPED PROPOSED IMPROVED TECHNICAL SPECIFICATIONS (ITS)

5.6 Reporting Requirements (continued)

## 5.7 High Radiation Area

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20:

5.7.1 High Radiation Areas with dose rates less than or equal to 1.0 rem/hour at 30 centimeters from the radiation source or from any surface penetrated by the radiation: Editorial

a. Each entrance or access point to such an area shall be barricaded (e.g., roped off) and conspicuously posted as a high radiation area, or be continuously guarded to prevent unauthorized personnel entry. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

b. Access to, and activities in, each such area shall be controlled by means of a Radiation Work Permit (RWP) or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures (e.g., radiation protection technicians) and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided they are following plant radiation protection procedures for entry to, exit from, and work in such areas. editorial

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously displays radiation dose rates in the area; or

2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the devices's dose alarm setpoint is reached, with a pre-set alarm setpoint, or

3. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area, or TSTF-258, R4

## 5.0 ADMINISTRATIVE CONTROLS

## 5.7 High Radiation Area

5.7.1 High Radiation Areas with dose rates less than or equal to 1.0 rem/hour at 30 centimeters from the radiation source or from any surface penetrated by the radiation: (continued)

Editorial

4. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

   (a) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

   TSTF-254,R4

   (b) Be under the surveillance, as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the mean to communicate with individuals in the area who are covered by such surveillance.

e. Except for individuals qualified in radiation protection procedures, or individuals escorted by personnel qualified in radiation protection procedures, entry into such areas shall be made only after dose rates in the area have been evaluated and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job breifing prior to entry into such areas. This dose rate determination, knowledge, and pre-job breifing does not require documentation prior to initial entry.

## 5.0 ADMINISTRATIVE CONTROLS

### 5.7 High Radiation Area

Editorial

5.7.2 High Radiation Areas with dose rates greater than 1.0 rem/hour at 30 centimeters from the radiation source or from any surface penetrated by the radiation, but less than 500 rads/hour at 1 meter from the radiation source or from any surface penetrated by the radiation:

a. Each entrance or access point to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked door or gate that prevents unauthorized entry, or be continuously guarded to prevent unauthorized personnel entry, and, in addition:

   1. All such door and gate keys shall be maintained under the administrative control of the shift supervisor, radiation protection manager, or designee.

   2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.

b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided they are following plant radiation protection procedures for entry to, exit from, and work in such areas.

TSTF-258, R4

d. Each individual or group entering such an area shall possess:

   1. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with a pre-set alarm setpoint, or

   2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area, or

## 5.0 ADMINISTRATIVE CONTROLS

### 5.7 High Radiation Area

5.7.2 High Radiation Areas with dose rates greater than 1.0 rem/hour at 30 centimeters from the radiation source or from any surface penetrated by the radiation, but less than 500 rads/hour at 1 meter from the radiation source or from any surface penetrated by the radiation: (continued)

3. A self reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

   (a) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

   (b) Be under the surveillance, as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area, or,

4. In those cases where options 2 and 3, above, are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle, a radiation monitoring device that continuously displays radiation dose rates in the area may be used.

e. Except for individuals qualified in radiation protection procedures, or individuals escorted by personnel qualified in radiation protection procedures, entry into such areas shall be made only after dose rates in the area have been evaluated and entry personnel are knowledgeable of them.

## 5.0 ADMINISTRATIVE CONTROLS

### 5.7 High Radiation Area

5.7.2 High Radiation Areas with dose rates greater than 1.0 rem/hour at 30 centimeters from the radiation source or from any surface penetrated by the radiation, but less than 500 rads/hour at 1 meter from the radiation source or from any surface penetrated by the radiation: (continued)

f. Such individual areas that are within a larger area where no enclosure exists for the purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate, nor conspicuously guarded, but shall be barricaded, conspicuously posted, and a clearly visible flashing light shall be activated at the area as a warning device.

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# JAFNPP

## IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

## CTS: 6.0

Administrative Controls

MARKUP OF CURRENT TECHNICAL SPECIFICATIONS (CTS)

DISCUSSION OF CHANGES (DOCs) TO THE CTS

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC) FOR LESS RESTRICTIVE CHANGES

# JAFNPP

## IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

**CTS: 6.0**

Administrative Controls

MARKUP OF CURRENT TECHNICAL SPECIFICATIONS (CTS)

CTS Chapter 6.0



see ITS: Section 5.3

## 6.3 PLANT STAFF QUALIFICATIONS

6.3.1 The minimum qualifications with regard to educational background and experience for plant staff positions shown in FSAR Figure 13.2-7 shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions; except for the Radiological and Environmental Services Manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

See ITS: Sections 5.2

6.3.2 The Shift Technical Advisor (STA) shall meet or exceed the minimum requirements of either Option 1 (Combined SRO/STA Position) or Option 2 (Continued use of STA Position), as defined in the Commission Policy Statement on Engineering Expertise on Shift, published in the October 28, 1985 Federal Register (50 FR 43621). When invoking Option 1, the STA role may be filled by the Shift Manager or Control Room Supervisor. (1)

6.3.3 Any deviations will be justified to the NRC prior to an individual's filling of one of these positions.

NOTE:

(1) The 13 individuals who hold SRO licenses, and have completed the FitzPatrick Advanced Technical Training Program prior to the issuance of License Amendment 111, shall be considered qualified as dual-role SRO/STAs.

## 6.4 RETRAINING AND REPLACEMENT TRAINING

A training program shall be maintained under the direction of the Training Manager to assure overall proficiency of the plant staff organization. It shall consist of both retraining and replacement training and shall meet or exceed the minimum requirements of Section 5.5 of ANSI N18.1-1971.

LA1

The retraining program shall not exceed periods two years in length with a curriculum designed to meet or exceed the requalification requirements of 10 CFR 55.59

## 6.5 REVIEW AND AUDIT

Review requirements are completed by using designated technical reviewers/qualified safety reviewer and two separate review committees. The Plant Operating Review Committee (PORC) is an onsite review group; the Safety Review Committee (SRC) is an independent offsite review and audit group.

LA2

### 6.5.0 REVIEW AND APPROVAL OF PROGRAMS AND PROCEDURES

6.5.0.1 The procedure review and approval process shall be controlled and implemented by administrative procedure(s).

6.5.0.2 Each program and procedure required by Specification 6.8 and other procedures that affect nuclear safety, and changes thereto, shall be reviewed by a minimum of two designated technical reviewers who are knowledgeable in the affected functional area.



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6.5.0.3 Designated technical reviewer(s) shall meet or exceed the qualifications described in Section 4 of ANSI N18.1-1971 for applicable positions, with the exclusion of the positions identified in Sections 4.3.2 and 4.5. Individuals whose positions are described in Sections 4.3.2 and 4.5 may qualify as designated technical reviewers provided they meet the qualifications described in other portions of Section 4.

6.5.0.4 The designated technical reviewer shall determine the need for cross-disciplinary reviews. Individuals performing cross-disciplinary reviews shall meet or exceed the qualifications described in Section 4 of ANSI N18.1-1971 for applicable positions.

6.5.0.5 Each program and procedure required by Specification 6.8 and other procedures that affect nuclear safety, and changes thereto, shall be reviewed from a safety perspective by a qualified safety reviewer. Safety and/or environmental impact evaluations, when required, shall be reviewed by PORC per Specification 6.5.1.6.a. 2

6.5.0.6 Nuclear safety related procedures and procedure changes shall be reviewed and approved, prior to implementation, by the appropriate member(s) of management.

6.5.1 <u>PLANT OPERATING REVIEW COMMITTEE (PORC)</u>

6.5.1.1 Function

PORC shall function to advise the Site Executive Officer on matters related to nuclear safety and environmental impact.

6.5.1.2 Membership

The Plant Operating Review Committee shall be composed of a Chairman, Vice-Chairmen and members designated in writing by the Site Executive Officer. Members shall collectively have responsibility in at least the areas of Operations, Maintenance, Radiological and Environmental Services, Engineering, Reactor Engineering, Instrumentation and Controls, Nuclear Licensing, and Quality Assurance.



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LA 2

6.5.1.3 Alternates

Alternative members shall be appointed in writing by the PORC Chairman to serve on a temporary basis.

6.5.1.4 Meeting Frequency

Meetings will be called by the Chairman as the occasions for review or investigation arise. Meetings will be no less frequent than once a month.

6.5.1.5 Quorum

A quorum of the PORC shall consist of the Chairman or a Vice-Chairman and five members including designated alternates. Vice-Chairmen may act as members when not acting as Chairman. A quorum shall contain no more than two alternates.

6.5.1.6 Responsibilities

The PORC shall be responsible for the:

a. Review of 10 CFR 50.59 safety and environmental impact evaluations associated with procedures and programs required by Specification 6.8, and changes thereto.

b. Review of proposed tests and experiments that affect nuclear safety.

c. Review of proposed changes to the Operating License and Technical Specifications.

d. Review of proposed changes or modifications to plant systems or equipment that affect nuclear safety.

e. Investigation of violations of the Technical Specifications. The PORC shall prepare and present a report covering the evaluations and recommendations to prevent recurrence to the Site Executive Officer, who will then forward the report to the Chief Nuclear Officer, the Director Regulatory Affairs and Special Projects, and to the Chairman of the Safety Review Committee.

f. Review of plant operations to detect potential safety hazards.

g. Performance of special reviews and/or investigations at the request of the Site Executive Officer.

h. Review of all reportable events.

i. Review of the Process Control Program and the Offsite Dose Calculation Manual (ODCM) and changes thereto.

6.5.1.3 Alternates

Alternative members shall be appointed in writing by the PORC Chairman to serve on a temporary basis.

6.5.1.4 Meeting Frequency

Meetings will be called by the Chairman as the occasions for review or investigation arise. Meetings will be no less frequent than once a month.

6.5.1.5 Quorum

A quorum of the PORC shall consist of the Chairman or a Vice-Chairman and five members including designated alternates. Vice-Chairmen may act as members when not acting as Chairman. A quorum shall contain no more than two alternates.

6.5.1.6 Responsibilities

The PORC shall be responsible for the:

a. Review of 10 CFR 50.59 safety and environmental impact evaluations associated with procedures and programs required by Specification 6.8, and changes thereto.

b. Review of proposed tests and experiments that affect nuclear safety.

c. Review of proposed changes to the Operating License and Technical Specifications.

d. Review of proposed changes or modifications to plant systems or equipment that affect nuclear safety.

e. Investigation of violations of the Technical Specifications. The PORC shall prepare and present a report covering the evaluations and recommendations to prevent recurrence to the Site Executive Officer, who will then forward the report to the Chief Nuclear Officer, the Director Regulatory Affairs and Special Projects, and to the Chairman of the Safety Review Committee.

f. Review of plant operations to detect potential safety hazards.

g. Performance of special reviews and/or investigations at the request of the Site Executive Officer.

h. Review of all reportable events.

i. Review of the Process Control Program and the Offsite Dose Calculation Manual (ODCM) and changes thereto.

j. Review the FitzPatrick Fire Protection Program and implementing procedures and changes thereto.

See JPTS-95-005

6.5.1.7- Authority -92

The PORC shall:

a. Recommend to the Site Executive Officer, approval or disapproval of those items reviewed under Specifications 6.5.1.6.a through 6.5.1.6.d.

b. Render determinations with regard to whether or not items considered under Specification 6.5.1.6.a through 6.5.1.6.e constitute an Unreviewed Safety Question as defined in 10 CFR 50.59.

c. In the event of a disagreement between the PORC and Site Executive Officer, notify the Chief Nuclear Officer and the SRC Chairman, or their designated alternates, within 24 hours and provide written notification by the next business day. The Site Executive Officer shall have responsibility for resolution of such disagreement pursuant to Section 6.1.

6.5.1.8 Records

Minutes of all meetings of the PORC shall be recorded and numbered. Copies will be retained in file. Copies will be forwarded to the Chairman of the SRC and the Chief Nuclear Officer.

6.5.1.9 Procedures

Conduct of the PORC and the mechanism for implementation of its responsibilities and authority are defined in the pertinent Administrative Procedures.

6.5.2 SAFETY REVIEW COMMITTEE (SRC)

FUNCTION

6.5.2.1 The SRC shall function to provide independent review and audit of designated activities in the areas of:

a. Nuclear power plant operations

b. Nuclear engineering

c. Chemistry and radiochemistry

d. Metallurgy

e. Instrumentation and control

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f. Radiological safety
g. Mechanical engineering
h. Electrical engineering
i. Administrative controls and quality assurance practices
j. Environment
k. Civil/Structural Engineering
l. Nuclear Licensing
m. Emergency Planning
n. Other appropriate fields associated with the unique characteristics of a nuclear power plant

LA2

CHARTER

6.5.2.2 The conduct of the SRC will be in accordance with a charter approved by the Chief Nuclear Officer. The charter will define the SRC's authority and establish the mechanism for carrying out its responsibilities.

MEMBERSHIP

6.5.2.3 The SRC shall be composed of at least six individuals including a Chairman and a Vice Chairman. Members shall be appointed by the Director Regulatory Affairs and Special Projects and approved by the Chief Nuclear Officer. SRC members and alternates shall have an academic degree in engineering or a physical science, or the equivalent, and shall have a minimum of five years technical experience in one or more areas listed in Section 6.5.2.1.

ALTERNATES

6.5.2.4 Alternates for the Chairman, Vice Chairman and members may be appointed in writing by the Director Regulatory Affairs and Special Projects and approved by the Chief Nuclear Officer.

CONSULTANTS

6.5.2.5 Consultants may be used as determined by the SRC Chairman and as provided for in the charter.

MEETING FREQUENCY

6.5.2.6 The SRC shall meet at least once per six months.

QUORUM

6.5.2.7 A quorum shall consist of at least a majority of the appointed individuals (or their alternates) and the Chairman (or the designated alternate). No more than two alternates may participate as SRC voting members at any one time. No more than a minority of the quorum shall have direct line responsibility for the operation of the plant.

REVIEW

6.5.2.8 The SRC shall review facility activities in accordance with the Quality Assurance Program, as described in Chapter 17 of the JAF FSAR.

AUDIT

6.5.2.9 Audits of facility activities shall be performed under the cognizance of the SRC and in accordance with the Quality Assurance Program, as described in Chapter 17 of the JAF FSAR.

LA2

CTS Chapter 6.0



AUTHORITY

6.5.2.10 The SRC shall advise the Chief Nuclear Officer on those areas of responsibility specified in Section 6.5.2.8 and 6.5.2.9.

LA2

B

Amendment No. ~~50, 60, 66, 78, 93, 94, 220, 222, 228~~, 251

CTS Chapter 6.0



RECORDS

6.5.2.11 Records will be maintained in accordance with ANSI 18.7-1972. The following shall be prepared and distributed as indicated below:

a. Minutes of each SRC meeting shall be prepared and forwarded to the Chief Nuclear Officer within 30 days after the date of the meeting.

b. Reports of reviews encompassed by Section 6.5.2.8 above shall be processed in accordance with the Quality Assurance Program, as described in Chapter 17 of the JAF FSAR.

c. Audit reports encompassed by Section 6.5.2.9 above, shall be processed in accordance with the Quality Assurance Program, as described in Chapter 17 of the JAF FSAR.

LA2

A1

LA3

A1

LA3

## 6.6 REPORTABLE EVENT ACTION

The following actions shall be taken for Reportable Events:

(A) The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

(B) Each Reportable Event shall be reviewed by the PORC, and the results of this review shall be submitted to the Chief Nuclear Officer, the Director Regulatory Affairs and Special Projects, and the Chairman of the SRC.

see ITS: Chapter 2.0

## 6.7 SAFETY LIMIT VIOLATION

(A) If a safety limit is exceeded, the reactor shall be shut down and reactor operation shall only be resumed in accordance with the provisions of 10 CFR 50.36, (c)(1)(i).

(B) An immediate report of each safety limit violation shall be made to the NRC by the Site Executive Officer. The Chief Nuclear Officer, the Director Regulatory Affairs and Special Projects, and the Chairman of the SRC will be notified within 24 hours.

(C) The PORC shall prepare a complete investigative report of each safety limit violation and include appropriate analysis and evaluation of: (l) applicable circumstances preceding the occurrence, (2) effects of the occurrence upon facility component systems or structures and (3) corrective action required to prevent recurrence. The Site Executive Officer shall forward this report to the Chief Nuclear Officer, the Director Regulatory Affairs and Special Projects, the Chairman of the SRC, and the NRC.

See ITS: Section 5.4

## 6.8 PROCEDURES

(A) Written procedures and administrative policies shall be established, implemented, and maintained that:

1. meet or exceed the requirements and recommendations of Section 5 of ANSI 18.7-1972 "Facility Administrative Policies and Procedures."
2. are recommended in Appendix A of Regulatory Guide 1.33, November 1972.
3. implement the Fire Protection Program.
4. include programs specified in Appendix B of the Radiological Effluent Technical Specifications, Section 7.2.

(B) Each procedure of Specification 6.8.(A), and changes thereto, shall be approved prior to implementation by the appropriate responsible member of management as specified in Specification 6.5.0.

(B) SPECIAL REPORTS

1. Fifteen copies of the Evaluation Report of the results of the first five years of performance of the non-destructive inspection listed in Table 4.6-1 of Technical Specifications 4.6.F, Structural Integrity, relating to the FitzPatrick in-service inspection program shall be submitted to the NRC, Director of Operating Reactors, within three months of the completion of the fifth year of the program.

2. DELETED

See ITS: Section 5.6

6.10 RECORD RETENTION

(A) The following records shall be retained for at least five years:

1. Records and logs of facility operation covering time intervals at each power level.
2. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.
3. All Reportable Events.
4. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.
5. Records of reactor tests and experiments.
6. Records of changes made to Operating Procedures.
7. Records of radioactive shipments.
8. Records of sealed source leak tests and results.
9. Records of annual physical inventory of all source material of record.

LA4

(B) The following records shall be retained for the duration of the Facility Operating License:

1. Records of any drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.
2. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.
3. Records of facility radiation and contamination surveys.
4. Records of radiation exposure for all individuals entering radiation control areas.
5. Records of gaseous and liquid radioactive material released to the environs.

LA4

6. Records of transient or operational cycles for those facility components identified in Table 6.10-1.

See ITS: Section 5.5

7. Records of training and qualification for current members of the plant staff.
8. Records of in-service inspections performed pursuant to these Technical Specifications.
9. Records of Quality Assurance activities required by the Quality Assurance Manual.
10. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.
11. Records of meetings of the PORC and the SRC.
12. Records for Environmental Qualification which are covered under the provisions of paragraph 6.15.
13. Records of the service life of all hydraulic and mechanical snubbers, whose failure could adversely affect any safety-related system, including the date at which the service life commences and associated installation and maintenance records as of the effective date of this ammendment.

LA4

## 6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared and adhered to for all plant operations. These procedures shall be formulated to maintain radiation exposures received during operation and maintenance as far below the limits specified in 10 CFR 20 as practicable. The procedures shall include planning, preparation, and training for operation and maintenance activities. They shall also include exposure allocation, radiation and contamination control techniques, and final debriefing.

Amendment No. ~~22~~, ~~37~~, 92
Order dated October 24, 1980

See ITS: Section 5.7

A2

### 6.12 INDUSTRIAL SECURITY PROGRAM

An industrial security program shall be maintained throughout the life of the plant in accordance with the provisions of the Plant Security Plan.

### 6.13 EMERGENCY PLAN

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### 6.14 FIRE PROTECTION PROGRAM

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6.15 ENVIRONMENTAL QUALIFICATION

A. By no later than June 30, 1982 all safety-related electrical equipment in the facility shall be qualified in accordance with the provisions of: Division of Operating Reactors "Guidelines for Evaluating Environmental Qualification of Class IE Electrical Equipment in Operating Reactors" (DOR Guidelines); or, NUREG-0588 "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment", December 1979. Copies of these documents are attached to Order for Modification of License DPR-59 dated October 24, 1980.

B. By no later than December 1, 1980, complete and auditible records must be available and maintained at a central location which describe the environmental qualification method used for all safety-related electrical equipment in sufficient detail to document the degree of compliance with the DOR Guidelines or NUREG-0588. Thereafter, such records should be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified.

A3

## 6.16 PROCESS CONTROL PROGRAM (PCP)

A. The PCP shall be a manual containing operational information concerning the solidification of radioactive wastes from liquid systems.

B. The PCP shall be maintained at the plant consistent with these Technical Specifications and with approved plant procedures.

LA5

C. Revisions of the PCP:

1. shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the revisions were made effective. This submittal shall contain:

   a. sufficiently detailed information to support the rationale for the revisions without benefit of additional information;

   b. a determination that the revision did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

   c. documentation that the revision has been reviewed and found acceptable by the PORC.

2. shall become effective upon issue following review and acceptance by the PORC.

## 6.17 OFFSITE DOSE CALCULATION MANUAL (ODCM)

A. The ODCM shall describe the methodology and parameters to be used in the calculation of offsite doses due to radioactive gaseous and liquid effluents and in the calculation of gaseous and effluents monitoring instrumentation alarm/trip setpoints consistent with the applicable LCOs contained in these Technical Specifications.

B. The ODCM shall be maintained at the plant and shall reflect accepted methodologies and calculational procedures.

See ITS: Section 5.5

CTS Chapter 6.0

See ITS: Section 5.5

C. Revisions of the ODCM:

1. shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the revisions were made effective. This submittal shall contain:

   a. sufficiently detailed information to support the rationale for the revisions without benefit of additional information (information submitted shall consist of revised pages of the ODCM, with each page numbered and provided with an approval and date box, together with appropriate evaluations justifying the revisions);

   b. a determination that the revisions will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

   c. documentation that the revisions have been reviewed and found acceptable by the PORC.

2. shall become effective upon issue following review and acceptance by the PORC.

TSTF-76, R1

6.18 MAJOR MODIFICATIONS TO RADIOACTIVE LIQUID, GASEOUS AND SOLID WASTE TREATMENT SYSTEMS*

LA7

A. Major modifications to radioactive waste systems (liquid, gaseous and solid):

1. shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the modification is completed and made operational. The discussion of each modification shall contain:

   a. a summary of the evaluation that led to the determination that the modification could be made in accordance with 10 CFR 50.59;

   b. sufficient information to support the reason for the modification without benefit of additional or supplemental information; and

   c. a description of the equipment, components and processes involved and the interfaces with other plant systems.

LA6

*The Authority may elect to submit the information called for in this Specification as part of the annual 10 CFR 50.59 Safety Evaluation Report.

2. The following evaluations shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report, where such evaluations are required to be performed in order to assure compliance with the requirements of 10 CFR 50.59:

   a. an evaluation of the modification, which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto.

   b. an evaluation of the modification, which shows expected maximum exposures to individuals in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the license application and amendments thereto; and

   c. a comparison of the predicted release of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the modifications are to be made.

LA6

CTS Chapter 6.0
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A4

27

CTS Chapter 6.0
Blank Pages

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A4

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Amendment No. ~~60~~, ~~76~~, ~~105~~, ~~121~~

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NOTE: Pages 262 through 264 were deleted by Amendment No. 37

The next page is Page 285

# JAFNPP

## IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

## CTS: 6.0

Administrative Controls

## DISCUSSION OF CHANGES (DOCs) TO THE CTS

## ADMINISTRATIVE CHANGES

A1 CTS 6.6.A contains requirements regarding notification and submittal of reports to the NRC pursuant to the requirements of 10 CFR 50.73. These reporting requirements are specified within the cited regulations and need not be repeated in the ITS. Since this change does not modify any technical requirements, it is administrative and has no adverse impact on safety.

A2 CTS 6.12 requires that an industrial security program be maintained throughout the life of the plant in accordance with the provisions of the Plant Security Plan. Security requirements are adequately addressed in 10 CFR 73.55, and need not be repeated in the ITS. Since this change does not modify any technical requirements, it is administrative and has no adverse impact on safety.

A3 CTS 6.15 requires that by June 30, 1982, all safety-related electrical equipment be environmentally qualified in accordance with the Division of Operating Reactors (DOR) Guidelines or NUREG-0588. It further requires that complete and auditable environmental qualification records be available and maintained at a central location by December 1, 1980. These requirements have been satisfactorily met. Environmental qualification requirements are adequately addressed in 10 CFR 50.49, and need not be repeated in the ITS. Since this change involves deletion of requirements that have been fulfilled, it is administrative and has no adverse impact on safety.

A4 The following intentionally blank CTS pages have been deleted: 254e, 254f, 257, 259, 260, and 262 through 284. This change is administrative, and has no adverse impact on safety.

## TECHNICAL CHANGES - MORE RESTRICTIVE

None

## TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA1 Details in CTS 6.4 which provide requirements for retraining and replacement training for the plant staff are not retained in the ITS and are being relocated to the UFSAR.

The requirements for training and qualification are adequately addressed in 10 CFR 50.120 and 10 CFR 55.59. Therefore, these details are not

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA1 (continued)

required to be in the ITS to provide adequate protection of the public health and safety, and relocation of these details is acceptable. Changes to the UFSAR will be controlled by the provisions of 10 CFR 50.59.

LA2 Details in CTS 6.5 which set forth requirements for review and approval of programs and procedures, and the review and audit responsibilities of the Plant Operating Review Committee (PORC) and Safety Review Committee (SRC) are not retained in ITS and are being relocated to the Quality Assurance Program description.

The requirements for the review and audit activities performed by the PORC and SRC are adequately addressed by ANSI N18.7, 10 CFR 50.54, 10 CFR 50.59, 10 CFR 50 Appendix B, and other applicable regulations and standards. Therefore, these details are not required to be in the ITS to provide adequate protection of the public health and safety, and relocation of these details is acceptable. Changes to the Quality Assurance Program description will be controlled by the provisions of 10 CFR 50.54(a). Additionally, NRC Administrative Letter 95-06 specifies that these details may be adequately addressed in the Quality Assurance Program.

LA3 Details in CTS 6.6(B) which set forth requirements for review and approval of reportable events are not retained in the ITS and are being relocated to the UFSAR.

The requirements for review of Reportable Events are adequately addressed in 10 CFR 50.73, 10 CFR 50.59, ANSI N18.7, and other applicable regulations and standards. In addition, given that these reviews and submittal of results are required following the event without a specified completion time, the relocated requirements are not necessary to ensure operation of the plant in a safe manner. Therefore, these details are not required to be in the ITS to provide adequate protection of the public health and safety, and relocation of these details is acceptable. Changes to the UFSAR will be controlled by the provisions of 10 CFR 50.59.

LA4 Details in CTS 6.10 which require that certain records be retained, are not retained in the ITS and are being relocated to The Quality Assurance Program description.

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA4 (continued)

The requirements for retention of records related to activities affecting quality are contained in 10 CFR 50 Appendix B, Criterion XVII and other sections of 10 CFR 50 that are applicable to JAFNPP. These record retention requirements provide a record of certain activities important to plant safety, but the records themselves do not assure safe operation of the plant since review of these records is a post-compliance review. Relocation of these CTS provisions to the "Quality Assurance Program - Operations" where they already exist assures adequate controls over record retention requirements. Therefore, these details are not required to be in the ITS to provide adequate protection of the public health and safety, and relocation of these details is acceptable. Changes to the Quality Assurance Program description will be controlled by the provisions of 10 CFR 50.54(a). Additionally, NRC Administrative Letter 95-06 specifies that these details may be adequately addressed in the Quality Assurance Program.

LA5 The details in CTS 6.16 which provide programmatic requirements for the processing of solid radioactive waste, the Process Control Program (PCP), are not retained in the ITS and are being relocated to the UFSAR.

The requirements for solid waste processing as implemented by the PCP are adequately addressed in 10 CFR 20, 10 CFR 61, and 10 CFR 71, and compliance with these provisions are requirements of the JAFNPP Operating License. Thus, relocation of these details does not affect the safe operation of the plant. Therefore, these details are not required to be in the ITS to provide adequate protection of the public health and safety, and relocation of these details is acceptable. Changes to the UFSAR will be controlled by the provisions of 10 CFR 50.59.

LA6 Details in CTS 6.18, associated with the reporting requirements of the major modifications to liquid, gaseous, and solid radwaste treatment systems are not retained in ITS and are being relocated to the Offsite Dose Calculation Manual (ODCM). This change is based on Generic Letter 89-01 which provides guidance on the removal of the Radiological Environmental Technical Specifications from the Technical Specifications. Therefore, the relocated details are not required to be in the ITS to provide adequate protection of the public health and safety, and relocation of these details is acceptable. Changes to the ODCM will be controlled in accordance with ITS 5.5.1, Offsite Dose Calculation Manual (ODCM).

## TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA7 Details in CTS 6.17.C.2 which set forth Plant Operating Review Committee (PORC) review and acceptance of changes to the Offsite Dose Calculation Manual (ODCM) are not retained in the ITS and are being relocated to the Quality Assurance Program. The PORC review and approval details are not required to be in the ITS to provide adequate protection of the public health and safety and therefore, the relocation of these details is acceptable. Quality Assurance Program changes will be controlled by the provisions of 10 CFR 50.54.

TSTF-76, R1

## TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

None

## TECHNICAL CHANGES - RELOCATIONS

None

# JAFNPP

## IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

**CTS: 6.0**

Administrative Controls

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC) FOR LESS RESTRICTIVE CHANGES

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 6.0 - ADMINISTRATIVE CONTROLS

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

There are no plant specific less restrictive changes for this Specification.

# MODIFIED/NEW RAI RESPONSES FOR ITS CHAPTER 5.0

Revision H changes to Chapter 5.0 RAI Responses

**5.5-02** CTS 4.7 & 4.11
DOC M.2
ITS/STS 5.5.8
JFD CLB3

In the STS markup section 5.5.8, the use of Regulatory Guide 1.52 and ASME N510-1989 has been replaced with insert 5.5.8-1 in which the basis for this deviation was provided as the Current Licensing Basis (CBL3). Part of this insert uses the term "significant" in reference to painting, fire, or chemical release in any ventilation zone communicating with the system. This term is not used in Regulatory Guide 1.52 or ASME N510-1989. It is understandable and reasonable in that this usage of the term "significant" is meant to define a threshold at which the test frequency should occur following painting, fire, or chemical releases. Conversely the term "significant" being used without specific definition does not really set any defined thresholds and it is left to individual for interpretation, be it the operator or the resident inspector. In addition, the use of this term is not in the (provided) current licensing basis and does not have an associated DOC. Recently this issue has been addressed in several correspondences between the NRC and the industry. In a letter from Mr. Jack N. Donahue of the NRC to Mr. Jerrold G. Deweese of Entergy Operations, Inc. dated September 11, 1997 the NRC offered guidance relevant to this particular issue.

**Comment:** Review and revise insert 5.5.8-1 as necessary to encompass a reasonable understanding of when this test should be performed. This review and revision should take in to account the previously mentioned letter and Regulatory Guide 1.52, Revision 2 and in addition, there should also be a DOC/JFD to explain this change. Another option would be simply to adopt in total Regulatory Guide 1.52, Revision 2 and ASME N510-1989 as provided in first the bracketed item in STS 5.5.8.

**JAFNPP Response:**

None provided.

**New Response:**

1. The Licensee will revise ITS 5.5.8 to address the reviewer comment. The term "significant" will be deleted. In place of the term "significant" the phrase "that could adversely affect the ability of the filter system to perform the intended function" will be added to HEPA filter testing requirements that are associated with painting, fire, or chemical release. (A similar phrase is contained CTS 4.7.B.1.c and CTS 4.11.A.2 which address charcoal adsorber testing requirements that are associated with painting, fire, or chemical release.) The addition of the phrase to the ITS HEPA filter test requirements, and the retention of the CTS phrase in the charcoal absorber testing requirements, is consistent with the guidance contained in the September 11, 1997 letter from J. N. Donahue (NRC) to J. G. Deweese (Entergy) referred to by the reviewer.

Revision H changes to Chapter 5.0 RAI Responses

2. Additional discussion of the retention of the CTS phrase concerning painting, fire, or chemical release that could affect the ability of the filter system to perform the intended function will be provided in DOC A13. Changes will also be made to DOC M2 to address addition of the phrase to the portions of ITS 5.5.8 that concern HEPA filter testing.

**5.5-03** CTS 4.7 & 4.11
DOC M.2
ITS/STS 5.5.8
JFD CLB3

In the STS markup section 5.5.8, the use of Regulatory Guide 1.52 and ASME N510-1989 has been replaced with insert 5.5.8-1 in which the basis for this deviation was provided as the Current Licensing Basis(CBL3). Assuming that insert 5.5.8-1 reflects the general intent of Regulatory Guide 1.52, Revision 2, several inconsistencies have been noted. In insert 5.5.8-1, the sentence that starts with "After each complete or partial...." should include the phrase "or removal of the charcoal absorber sample." at the end to make it consistent with Regulatory Guide 1.52. In addition, the sentence pertaining to 5.5.8.c that starts with "After any structural maintenance..." includes the term "HEPA filter" which is actually not necessary in this particular section.

**Comment:** Revise submittal to address issues as discussed above or provide discussion to justify why these changes are not necessary.

**JAFNPP Response:**

None provided.

**New Response:**

1. The Licensee will revise ITS 5.5.8 to address the reviewers comments.

2. Changes will be made to the CTS markup and ITS 5.5.8.c to require charcoal adsorber penetration and bypass testing "after removal of a charcoal sample." In addition, reference to HEPA filter testing will be removed from the DOC M2 discussion concerning additions to CTS that are in ITS 5.5.8.c since ITS 5.5.8.c does not address HEPA filter testing.

**5.5-04** CTS 4.7 & 4.11
DOC VARIOUS
ITS/STS 5.5.8
JFD VARIOUS

In the STS markup section 5.5.8, references of the ASME N510-1989 standard have been deleted in several places and kept in section 5.5.8.e. It is unclear why the ASME N510-1989 standard is not used consistently throughout section 5.5.8.

**Comment:** Provide response for inconsistency in ASME N510-1989 standard usage as stated above.

**JAFNPP Response:**

None provided.

**New Response:**

1. The Licensee will revise ITS 5.5.8.a and ITS 5.5.8.b by indicating the revision of ASME N510 currently specified in plant procedures associated with testing of HEPA and charcoal absorber filters. That is, ASME N510-1980. The Standby Gas Treatment (SGT) and Control Room Emergency Ventilation Air Supply (CREVAS) Systems were designed and constructed to ASME N509-1976, Nuclear Power Plant Air-Cleaning Units and Components, and ASME N510-1980 is the appropriate testing standard for the SGT and CREVAS System HEPA and charcoal adsorber filter systems. A new JFD will be added to provide this information.

2. ITS 5.5.8.d continues with the reference to ASME N510 being deleted because as noted in JFD X4, ASME N510 does not address filter pressure drop testing.

3. ITS 5.5.8.e continues to refer to ASME N510-1975 because that is the version currently specified in plant procedures.

**5.5-05** CTS 4.7 & 4.11
DOC NA
ITS/STS insert 5.5.8-2(5.5.8.a), insert 5.5.8-3(5.5.8.b), insert 5.5.8-5(5.5.8.b)
JFD PA9

In the STS markup section 5.5.8, insert 5.5.8-2, insert 5.5.8-3, and insert 5.5.8-5 have been provided for the bracketed items in 5.5.8.a, 5.5.8.b, and 5.5.8.e respectively. The numerical values in the column "Penetration" from insert 5..5.8-2, 5.5.8-3, and 5.5.8-5 do not appear to be consistent with Regulatory Guide 1.52 and it is not clear if these value are actually in the current licensing basis.

**Comment:** Provide response for inconsistency in Regulatory Guide 1.52 usage or provide supplemental current licensing basis information to support the numerical values as stated above.

**JAFNPP Response:**

None provided.

**New Response:**

1. The HEPA filter bypass values contained in ITS 5.5.8.a are current licensing basis values as shown in CTS 4.7.B.1.b.(1) and 4.11.A.1.b for the Standby Gas Treatment (SGT) and Control Room Emergency Ventilation Air Supply (CREVAS) Systems respectively. Refer to CTS markup pages 16 (for SGT) and 19 (for CREVAS).

2. The charcoal adsorber bypass values contained in ITS 5.5.8.b are current licensing basis values as shown in CTS 4.7.B.1.b.(2) and 4.11.A.1.c for the Standby Gas Treatment (SGT) and Control Room Emergency Ventilation Air Supply (CREVAS) Systems respectively. Refer to CTS markup pages 16 (for SGT) and 19 (for CREVAS).

3. The methyl iodide penetration values for charcoal adsorber sample laboratory analysis contained in ITS 5.5.8.c are current licensing basis values as shown in CTS 4.7.B.1.c.(1) and 4.11.A.2.(1) for the Standby Gas Treatment and CREVAS Systems respectively. Refer to CTS markup pages 16 (for SGT) and 19 (for CREVAS). Note that these values were changed by CTS Amendment 269 which reflected Generic Letter 99-02.

4. The specific reference to Section C.6.b of Regulatory Guide 1.52, Revision 2, contained in ITS 5.5.8.c is based on the current licensing basis. Refer to CTS markup at 4.7.B.1.c.(1) and 4.11.A.2.(1) on CTS markup pages 16 and 19 for SGT and CREVAS respectively. The specific references to Sections C.6.a, C.6.b, C.6.c, and C.6.d of Regulatory Guide 1.52, Revision 2 referred to in ITS 5.5.8.a and ITS 5.5.8.b were added for clarification as noted in JFD PA8.

**5.5-06** CTS 4.7 & 4.11
DOC NA
ITS/STS 5.5.8.d
JFD X4

In the STS markup section 5.5.8.d the term "in accordance with [Regulatory Guide....." has been deleted. The JFD X4 appears to adequately justify the change but this test needs to been done in accordance with some procedure, guide, or standard.

**Comment:** Review and revise issue mentioned above such that this test will be administered in accordance with proper regulatory procedures. This situation would appear to be a "Generic Issue" that would warrant forwarding to the "Technical Specification Task Force" (TSTF) Owners Group(s) for consideration.

**JAFNPP Response:**

None provided.

**New Response:**

1. Filter pressure drop testing is conducted in accordance with approved plant procedures. The pressure drop limits contained in ITS 5.5.8.d (and the plant procedures) are based on filter manufacturer specifications.

2. The Licensee plans to bring the apparent need for a generic change to NUREG Section 5.5.8.d to the attention of the BWROG Technical Specification Issues Coordinating Committee at the next scheduled committee meeting in July, 2001.

**5.5-07** CTS RETS 2.5
STS 5.5.10
ITS 5.5.9
JFD None

CTS RETS 2.5 contains the statement "the amount that would result in concentrations less than...." There is indication in the CTS markup that this statement will be relocated or deleted. The STS 5.5.10 markup deletes this statement and replaces it with the statement ".. Or equal to 10 curies..." CLB5 states that this change reflects JAFNPP's current requirements. For this DOC to be true, both previously mentioned statements would have to be included in the ITS.

**Comment:** Revise the submittal to either include both previously mentioned wording in the ITS or provide less restrictive documentation to justify this change.

**JAFNPP Response:**

1. The Authority will revise the submittal to address deletion of phrase.

**Revised Response:**

1. The License will revise CTS RETS 2.5 on CTS markup page 21 and will revise DOC A10 to provide discussion of the changes.

2. ITS 5.5.9.b will also be revised to express the limits of radioactivity contained in unprotected tanks to less than the quantity that would result in a concentration that is 10 times the concentration values in Appendix B, Table 2, Column 2 of 10 CFR 20.1001-20.2402 (excluding Tritium and dissolved or entrained noble gases) at the nearest potable water supply and the nearest surface water supply beyond the site boundary in the event of uncontrolled release of the tank contents. These changes make the limits in ITS 5.5.9.b consistent with the liquid release limits in ITS 5.5.4.b.. JFD X8 will be added for the changes to ITS 5.5.9.b.

Revision H changes to Chapter 5.0 RAI Responses

**5.6-01** CTS 6.9.A.4.d
STS/ITS 5.6.5.d

Last several words in paragraph CTS section 6.9.A.4.d have been deleted (omitted) with no discussion of change.

**Comment:** Noting that this change (deletion) is consistent with NUREG-1433. either provide DOC to justify change or retain original CTS wording.

**JAFNPP Response:**

None provided.

**New Response:**

1. The Licensee will revise the CTS markup for CTS 6.9.A.4.d to show deletion of the COLR distribution details that are a duplication of the requirements of 10 CFR 50.4 and add a new A DOC for the changes.

**5.7-01** CTS 6.11
STS 5.7
TSTF-258 R.4 sections 5.7.1.4.i, 5.7.1.4.ii, 5.7.2.a, 5.7.2.e.5.7.2.f
ITS 5.7.1.d.4, 5.7.1.e, 5.7.2.a, 5.7.2.e, 5.7.2.f
JFD X1

In the proposed section 5.7 change, JFD X1 references a letter from the NRC to the Owners Groups dated 7/28/95. This letter eventually evolved into what is now the NRC approved TSTF-258, R.4. The latest revision (R.4) of this generic issue was approved by the NRC on 4/99 and the TSTF (Owners Groups) were notified of deposition (approved) via letter dated 6/29/99. In this proposed section 5.7 submittal, the sections as referenced above are not consistent with the NRC approved TSTF-258 R.4.

**Comment:** If adopting TSTF-258 R.4, revise sections (as mentioned above) for consistency or provide justification for changes. If not adopting TSTF-258 R.4, provide additional JFD(s) (explicit and technical) to explain the differences and in addition, to justify why you are not adopting the NUREG-1433 R.1 or the TSTF-258 R.4 (which in a few months will be in the standard NUREG-1433 R.2)

**JAFNPP Response:**

1. The Authority will revise the submittal to reflect withdrawal of TSTF-86 and adopt the changes in TSTF-258, Revision 4.

**Revised Response:**

1. The Licensee will revise ITS 5.7 by adopting TSTF-258, Revision 4, with the following exceptions:

   a. The changes to ITS 5.7 that are contained in TSTF-258, Revision 4 will be revised (at ITS 5.7.1.a and ITS 5.7.2.a) to allow the continuous guarding of a high radiation area entrance or access point (in lieu of barricading or providing a locked door or gate) to address the potential for temporary or transient high radiation areas, broken door or gate locks, and other similar circumstances. Allowing a high radiation area entrance or access point to be continuously guarded is necessary to ensure positive control over personnel entry while maintaining compliance with ITS 5.7.1.a and ITS 5.7.2.a, during circumstances similar to those described.

   b. Several editorial changes will be made to the changes to ITS 5.7 that are contained in TSTF-258, Revision 4. These changes address inconsistencies with respect to the Writers' Guide for Restructured Technical Specifications or provide clarification.

# JAFNPP

## IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

## CTS: CTS RETS 2.1

Liquid Effluent Monitors

MARKUP OF CURRENT TECHNICAL SPECIFICATIONS (CTS)

DISCUSSION OF CHANGES (DOCs) TO THE CTS

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC) FOR LESS RESTRICTIVE CHANGES

# JAFNPP

## IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

## ITS: CTS RETS 2.1

Liquid Effluent Monitors

MARKUP OF CURRENT TECHNICAL SPECIFICATIONS (CTS)

| LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS |
|---|---|

## 2.0 LIQUID EFFLUENTS

### 2.1 LIQUID EFFLUENT MONITORS

**Applicability**

Applies to instrumentation required for monitoring radioactive liquid effluent discharges to the environment as specified in Table 2.1-1.

**Objective**

To assure that radioactive liquid effluent discharges are properly monitored and recorded during release.

**Specifications**

a. The limiting conditions for operation of the instruments that monitor radioactive liquid effluents are given in Table 2.1-1. With a radioactive liquid effluent monitoring instrumentation channel alarm/trip set point less conservative than required by the ODCM, without delay suspend the release of radioactive liquid effluents monitored by the affected channel, or declare the channel inoperable, or change the set point so it is acceptably conservative.

b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels operable, take the action shown in Table 2.1-1. Take corrective actions to return the instruments to operable

### 2.1 LIQUID EFFLUENT MONITORS

**Applicability**

Applies to instrumentation for monitoring radioactive liquid effluent discharges.

**Objective**

To ensure that instrumentation required for radioactive liquid effluent discharges are maintained and calibrated.

**Specifications**

a. The alarm/trip set points of these channels shall be determined and adjusted in accordance with the methodology and parameters in the Offsite Dose Calculation Manual (ODCM).

b. The surveillance requirements for the radioactive liquid effluent monitoring instrumentation is shown on Table 3.10-2.

R1

| LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS |
| --- | --- |

status within 30 days and, if unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

<u>NOTE:</u> This reporting requirement does not apply to instruments which would not have been required to be operable during the 30 day period.

CTS RETS 2.1

Table 2.1-1

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

| Instrument | Minimum Channels Operable | Action |
|---|---|---|
| Gross radioactivity monitors providing alarm and automatic termination of release | | |
| Liquid radwaste effluent line | 1 | (a) |
| Gross beta or gamma radioactivity monitors providing alarm but not providing automatic termination of release | | |
| Service water system effluent line | 1 | (b) |
| Flow rate measurement devices | | |
| Liquid radwaste effluent line | 1 | (c) |

NOTES FOR TABLE 2.1-1

(a) With the number of operable channels less than the required minimum number, effluent releases may continue provided that prior to initiating a release:

a. Two independent samples are analyzed;

b. Two technically qualified members of the facility staff verify the discharge line valving;

Otherwise, suspend release of radioactive effluents via this pathway.

(b) With the number operable of channels less than the required minimum number, effluent releases in this pathway may continue provided that, at least once per 12 hours, grab samples are collected and analyzed for principal gamma emitters at a limit of detection of at least $5x10^{-7}$ microcuries/ml. The principal gamma emitters for which the LLD specification applies exclusively are described in Note (c) to Table 2.3-1.

(c) With the number of operable channels less than the required minimum number, effluent releases via this pathway may continue provided the flow rate is estimated at least once per four hours during actual releases. Pump curves or tank level decreases generated in situ may be used to estimate flow.

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TABLE 3.10-2

MINIMUM TEST AND CALIBRATION FREQUENCY FOR RADIATION MONITORING SYSTEMS[a]

| | Instrument Channels | Instrument Check[b] | Instrument Channel Functional Test[i] | Instrument Channel Calibration | Logic System Function Test[(i)(h)] | |
|---|---|---|---|---|---|---|
| (1) | Main Stack Exhaust Monitors and Recorders | Daily | Quarterly | Quarterly | - - | See CTS RETS 3.1 |
| (2) | Refuel Area Exhaust Monitors and Recorders | Daily | Quarterly | Quarterly | - - | See ITS: 3.3.6.2 |
| (3) | Reactor Building Area Exhaust Monitors, Recorders, and Isolation | Daily | Quarterly | Quarterly | Once per 24 Months | |
| (4) | Turbine Building Exhaust Monitors and Recorders | Daily | Quarterly | Quarterly | - - | |
| (5) | Radwaste Building Exhaust Monitors and Recorders | Daily | Quarterly | Quarterly | - - | See ITS: 3.7.5 |
| (6) | SJAE Radiation Monitors/Offgas Line Isolation | Daily | Quarterly | Quarterly | Once per 24 Months | |
| (7) | Main Control Room Ventilation Monitor | Daily | Quarterly | Quarterly | - - | See ITS: 3.3.7.1 |
| (8) | Mechanical Vacuum Pump Isolation[g] | - - | - - | - - | Once per 24 Months | See ITS: 3.3.7.2 |
| (9) | Liquid Radwaste Discharge Monitor/ Isolation[(c)(d)(e)(f)] | Daily When Discharging | Quarterly | Quarterly | Once per 24 Months | |
| (10) | Liquid Radwaste Discharge Flow Rate Measuring Devices[d] | Daily | Quarterly | Once per 18 Months | - - | |
| (11) | Liquid Radwaste Discharge Radioactivity Recorder[d] | Daily | Quarterly | Once per 18 Months | - - | |
| (12) | Normal Service Water Effluent | Daily | Quarterly | Quarterly | - - | |
| (13) | SBGTS Actuation | - - | - - | - - | Once per 24 Months | See ITS: 3.3.6.2 3.6.4.3 |

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NOTES FOR TABLE 3.10-2

(a) Functional tests, calibrations and instrument checks need not be performed when these instruments are not required to be operable or are tripped.

(b) Instrument checks shall be performed at least once per day during these periods when the instruments are required to be operable.

R1

(c) A source check shall be performed prior to each release.

(d) Liquid radwaste effluent line instrumentation surveillance requirements need not be performed when the instruments are not required as the result of the discharge path not being utilized.

(e) An instrument channel calibration shall be performed with known radioactive sources standardized on plant equipment which has been calibrated with NBS traceable standards.

R1

(f) Simulated automatic actuation shall be performed once per 24 months. Where possible, all logic system functional tests will be performed using the test jacks.

R1

(g) Refer to Appendix A for instrument channel functional test and instrument channel calibration requirements (Table 4.2-1). These requirements are performed as part of main steam high radiation monitor surveillances.

see ITS: 3.3.7.2

(h) The logic system functional tests shall include a calibration of time delay relays and timers necessary for proper functioning of the trip systems.

(i) This instrumentation is excepted from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel. These instrument channels will be calibrated using simulated electrical signals once every three months.

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# JAFNPP

## IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

## ITS: CTS RETS 2.1

Liquid Effluent Monitors

DISCUSSION OF CHANGES (DOCs) TO THE CTS

DISCUSSION OF CHANGES
CTS RETS: 2.1 - LIQUID EFFLUENT MONITORS

ADMINISTRATIVE CHANGES

None

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

None

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

None

TECHNICAL CHANGES - RELOCATIONS

R1 CTS RETS 2.1 and the associated Surveillance Requirements of CTS RETS Table 3.10-2 contain requirements for the operability of the liquid effluent monitors. CTS RETS 2.1 ensures that radioactive liquid effluent releases are properly monitored and recorded during release and that liquid effluent monitor alarms and trip setpoints are established in accordance with the Offsite Dose Calculation Manual (ODCM). The alarm/trip setpoints are established to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR 20 and 10 CFR 50, Appendix I.

The radioactive liquid effluent monitors are used to show conformance to the discharge limits of 10 CFR 20 and 10 CFR 50, Appendix I. This instrumentation provides a continuous check on the release of radioactive liquid effluent from the liquid effluent flow paths. Plant Design Basis Accident (DBA) analyses do not assume any action, either initiated by, or resulting from information provided by radioactive effluent monitors.

Administrative controls are included in the Technical Specifications to ensure continued compliance with the applicable regulatory requirements. ITS 5.5.4, "Radioactive Effluent Controls Program" and ITS 5.5.1, "ODCM" contain requirements to ensure that all liquid effluents meet the limits contained in applicable regulations and future changes to the ODCM will be reviewed to ensure that such changes will "maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190,

TECHNICAL CHANGES - RELOCATIONS

R1 (continued)

10 CFR 50.36a, and 10 CFR 50, Appendix I and do not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations."

CTS RETS 2.1 does not identify a parameter which is an initial condition or assumption for a DBA or transient, identify a significant abnormal degradation of the reactor coolant pressure boundary, mitigate a design basis event and is not a structure system or component which operating experience or PRA has shown to be significant to public health and safety.

Therefore, CTS RETS 2.1 does not satisfy the criteria of 10 CFR 50.36(c)(2)(ii) as documented in the Application of Selection Criteria to the JAFNPP Technical Specifications and will be relocated to the ODCM. Changes to the ODCM will be controlled by the provisions of the ODCM change control process described in Chapter 5 of the ITS. This change is consistent with Generic Letter 89-01 for removal of Radiological Effluent Technical Specification (RETS) and relocation to the ODCM.

# JAFNPP

## IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

## ITS: CTS RETS 2.1

Liquid Effluent Monitors

## NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC) FOR LESS RESTRICTIVE CHANGES

NO SIGNIFICANT HAZARDS CONSIDERATIONS
CTS RETS: 2.1 - LIQUID EFFLUENT MONITORS

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

There are no plant specific less restrictive changes identified for this Specification.