October 16, 1986

Docket No. 50-389

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Dear Mr. Woody:	BWright	CLiang

The Commission has issued the enclosed Amendment No. 16 to Facility Operating License No. NPF-16 for the St. Lucie Plant, Unit No. 2. This amendment consists of changes to the Technical Specifications in response to your application dated July 15, 1986 as supplemented by two letters dated September 4, 1986 and one letter dated October 10, 1986.

This amendment changes the Reactor Coolant System Pressure/Temperature (P/T) limit figures to be effective up to four effective full power years of operation. The amendment changes the technical specifications dealing with overpressure protection systems because they are linked with the new P/T limit figures. The amendment also adds the shutdown cooling system relief valves as overpressure protection devices. The applicable bases sections are changed to reflect the above changes.

A copy of the related Safety Evaluation is also enclosed. The notice of issuance will be included in the Commission's next bi-weekly Federal Register notice.

Sincerely,

/s/

E. G. Tourigny, Project Manager
PWR Project Directorate #8
Division of PWR Licensing-B

Enclosures:
1. Amendment No. 16 to NPF-16
2. Safety Evaluation

cc w/enclosures:
See next page

PBD#8	PBD#8	FOB	OGC-Bethesda	PWR#8
PMKreutzer	ETourigny	WRegan	No legal objection McGurren	AThadani
10/14/86	10/14/86	10/16/86	10/16/86	10/16/86

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P PDR

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St. Lucie Plant

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UNITED STATES
NUCLEAR REGULATORY COMMIS...ON
WASHINGTON, D. C. 20555

FLORIDA POWER & LIGHT COMPANY

ORLANDO UTILITIES COMMISSION OF

THE CITY OF ORLANDO, FLORIDA

AND

FLORIDA MUNICIPAL POWER AGENCY

DOCKET NO. 50-389

ST. LUCIE PLANT UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 16
License No. NPF-16

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Florida Power & Light Company, et al. (the licensee), dated July 15, 1986 as supplemented by two letters dated September 4, 1986 and one letter dated October 10, 1986, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, Facility Operating License No. NPF-16 is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and by amending paragraph 2.C.2 to read as follows:

   2. Technical Specifications

      The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 16 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Ashok C. Thadani, Director
PWR Project Directorate #8
Division of PWR Licensing-B

Attachment:
Changes to the Technical Specifications

Date of Issuance: October 16, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. _____

TO FACILITY OPERATING LICENSE NO. NPF-16

DOCKET NO. 50-389

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Remove Pages	Insert Pages
VI	VI
1-4	1-4
3/4 4-3	3/4 4-3
3/4 4-5	3/4 4-5
3/4 4-29	3/4 4-29
3/4 4-30	3/4 4-30
3/4 4-31	3/4 4-31
3/4 4-32	3/4 4-32
3/4 4-33	3/4 4-33
3/4 4-34	3/4 4-34
3/4 4-35	3/4 4-35
3/4 4-36	3/4 4-36
3/4 5-7	3/4 5-7
B 3/4 4-1	B 3/4 4-1
B 3/4 4-8	B 3/4 4-8
B 3/4 4-11	B 3/4 4-11
B 3/4 5-2	B 3/4 5-2

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, " Calculation of Distance Factors for Power and Test Reactor Sites."

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.11 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

ENGINEERED SAFETY FEATURES RESPONSE TIME

1.12 The ENGINEERED SAFETY FEATURES RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

FREQUENCY NOTATION

1.13 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

GASEOUS RADWASTE TREATMENT SYSTEM

1.14 A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

1.15 IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured, and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor Coolant System leakage through a steam generator to the secondary system.

1.16 LOW TEMPERATURE RCS OVERPRESSURE PROTECTION RANGE

The LOW TEMPERATURE RCS OVERPRESSURE PROTECTIVE RANGE is that operating condition when (1) the cold leg temperature is $\leq$ 286°F during cooldown and $\leq$ 295°F during heatup and (2) the Reactor Coolant System has pressure boundary integrity. The Reactor Coolant System does not have pressure boundary integrity when the Reactor Coolant System is open to containment and the minimum area of the Reactor Coolant System opening is greater than 3.58 square inches.

MEMBER(S) OF THE PUBLIC

1.17 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the licensee, its contractors or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational or other purposes not associated with the plant.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.18 The OFFSITE DOSE CALCULATION MANUAL shall contain the current methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and shall include the Radiological Environmental Monitoring Sample point locations.

OPERABLE - OPERABILITY

1.19 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL MODE - MODE

1.20 An OPERATIONAL MODE (i.e. MODE) shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.2.

PHYSICS TESTS

1.21 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and (1) described in Chapter 14.0 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 At least two of the loop(s)/train(s) listed below shall be OPERABLE and at least one Reactor Coolant and/or shutdown cooling loops shall be in operation.*

a. Reactor Coolant Loop 2A and its associated steam generator and at least one associated Reactor Coolant pump,**

b. Reactor Coolant Loop 2B and its associated steam generator and at least one associated Reactor Coolant pump,**

c. Shutdown Cooling Train 2A,

d. Shutdown Cooling Train 2B.

APPLICABILITY: MODE 4.

ACTION:

a. With less than the above required Reactor Coolant and/or shutdown cooling loops OPERABLE, immediately initiate corective action to return the required loops to OPERABLE status as soon as possible; if the remaining OPERABLE loop is a shutdown cooling loop, be in COLD SHUTDOWN within 30 hours.

b. With no Reactor Coolant or shutdown cooling loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.

*All Reactor Coolant pumps and shutdown cooling pumps may be de-energized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

**A Reactor Coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 295°F during heatup or 286°F during cooldown unless the secondary water temperature of each steam generator is less than 40 °F above each of the Reactor Coolant System cold leg temperatures.

REACTOR COOLANT SYSTEM

HOT SHUTDOWN

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required Reactor Coolant pump(s), if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.3.2 The required steam generator(s) shall be determined OPERABLE by verifying the secondary side water level to be $\geq$ 10% indicated narrow range level at least once per 12 hours.

4.4.1.3.3 At least one Reactor Coolant or shutdown cooling loop shall be verified to be in operation and circulating Reactor Coolant at least once per 12 hours.

REACTOR COOLANT SYSTEM

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.1 At least one shutdown cooling loop shall be OPERABLE and in operation*, and either:

a. One additional shutdown cooling loop shall be OPERABLE#, or

b. The secondary side water level of at least two steam generators shall be greater than 10% indicated narrow range level.

APPLICABILITY: MODE 5 with Reactor Coolant loops filled##.

ACTION:

a. With one of the shutdown cooling loops inoperable and with less than the required steam generator level, immediately initiate corrective action to return the inoperable shutdown cooling loop to OPERABLE status or to restore the required steam generator level as soon as possible.

b. With no shutdown cooling loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required shutdown cooling loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4.1.1 The secondary side water level of at least two steam generators when required shall be determined to be within limits at least once per 12 hours.

4.4.1.4.1.2 At least one shutdown cooling loop shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

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*The shutdown cooling pump may be de-energized for up to 1 hour provided 1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and 2) core outlet temperature is maintained at least 10°F below saturation temperature.

#One shutdown cooling loop may be inoperable for up to 2 hours for surveillance testing provided the other shutdown cooling loop is OPERABLE and in operation.

##A Reactor Coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 295°F during heatup or 286°F during cooldown unless the secondary water temperature of each steam generator is less than 40 °F above each of the Reactor Coolant System cold leg temperatures.

COLD SHUTDOWN - LOOPS NOT FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.2 Two shutdown cooling loops shall be OPERABLE# and at least one shutdown cooling loop shall be in operation.*

APPLICABILITY: MODE 5 with reactor coolant loops not filled.

ACTION:

a. With less than the above required loops OPERABLE, within 1 hour initiate corrective action to return the required loops to OPERABLE status as soon as possible.

b. With no shutdown cooling loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and within 1 hour initiate corrective action to return the required shutdown cooling loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4.2 At least one shutdown cooling loop shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

#One shutdown cooling loop may be inoperable for up to 2 hours for surveillance testing provided the other shutdown cooling loop is OPERABLE and in operation.

*The shutdown cooling pump may be deenergized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

REACTOR COOLANT SYSTEM

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2 and 3.4-3 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operations or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psia, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, at the intervals required by 10 CFR 50 Appendix H in accordance with the schedule in Table 4.4-5. The results of these examinations shall be used to update Figures 3.4-2 and 3.4-3.

FIGURE 3.4-2
ST. LUCIE-2 P/T LIMITS, 4 EFPY
HEATUP AND CORE CRITICAL

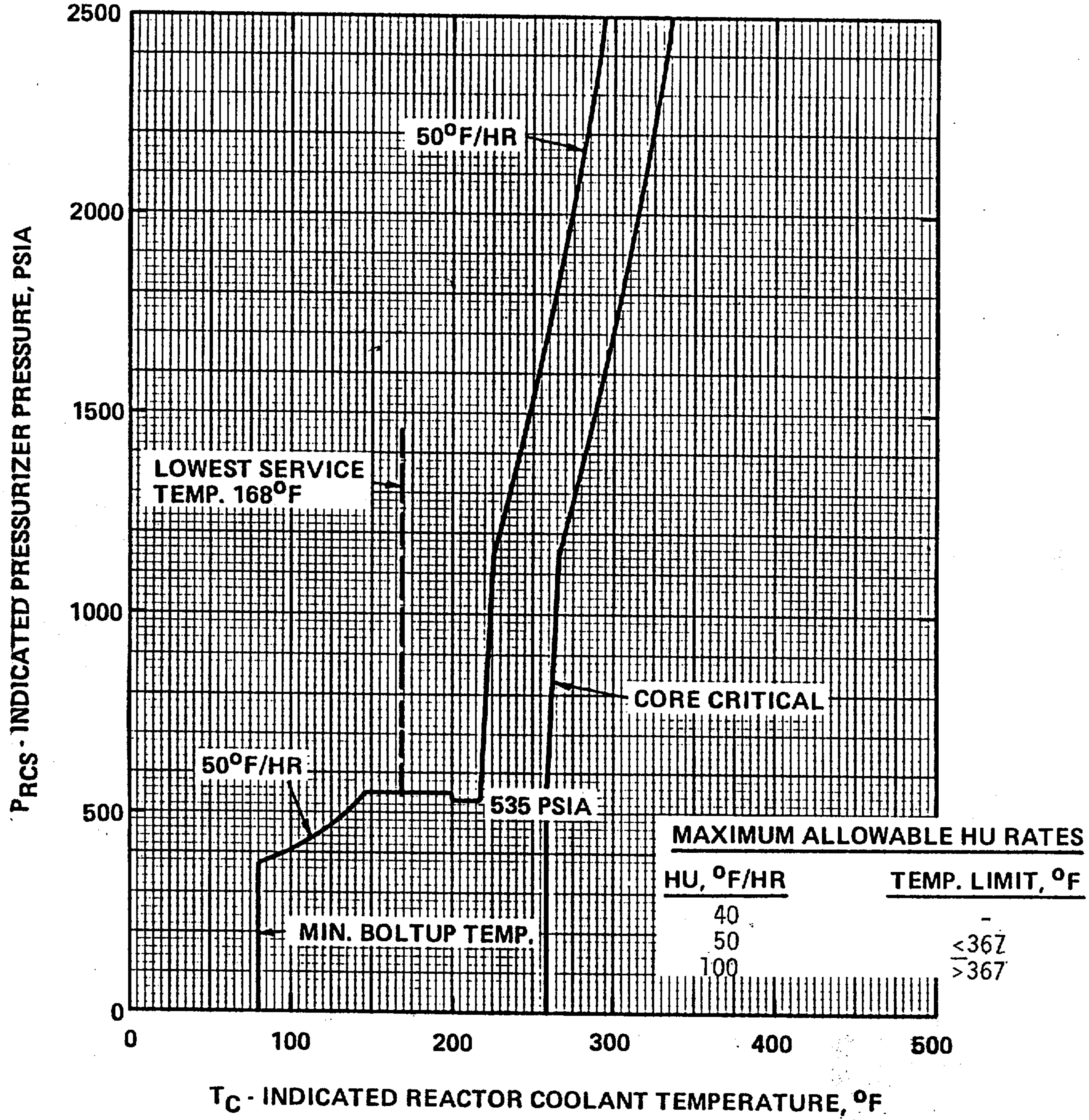

FIGURE 3.4-3
ST. LUCIE-2 P/T LIMITS, 4 EFPY
COOLDOWN AND INSERVICE TEST

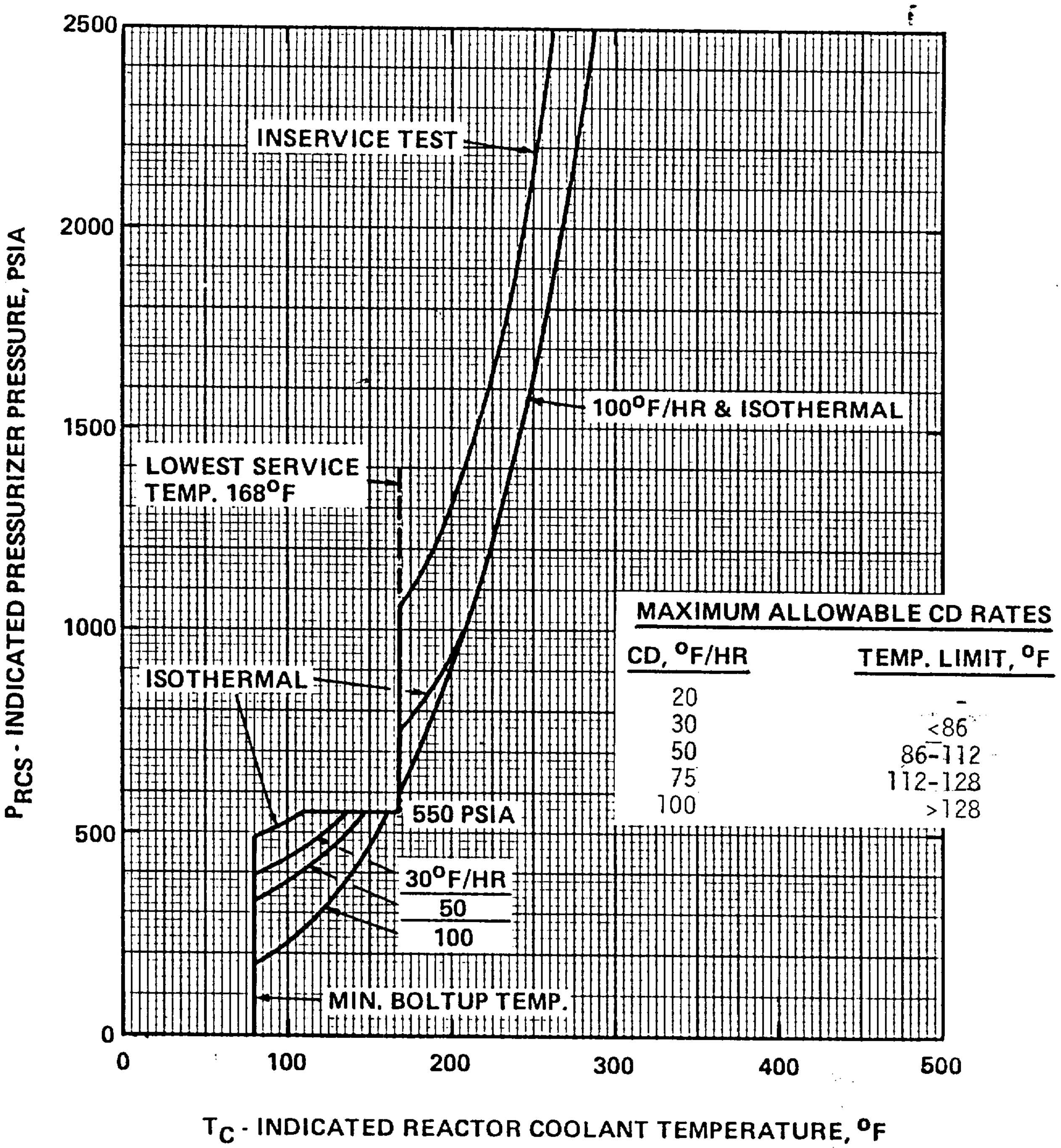

TABLE 4.4-5

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWAL SCHEDULE

CAPSULE NUMER	VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME (EFPY)
1	83°	1.15	1.0
2	97°	1.15	24.0
3	104°	1.15	STANDBY
4	263°	1.15	12.0
5	277°	1.15	STANDBY
6	284°	1.15	STANDBY

PRESSURIZER HEATUP/COOLDOWN LIMITS

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

a. A maximum heatup of 100°F in any 1-hour period, and

b. A maximum cooldown of 200°F in any 1-hour period.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.2 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown.

REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.3 At least one of the following overpressure protection systems shall be OPERABLE:

a. Two power-operated relief valves (PORVs), each with a lift setting of less than or equal to 470 psia, or

b. Two shutdown cooling relief valves (SDCRVs) with a lift setting of less than or equal to 350 psia, or

c. The Reactor Coolant System depressurized with an RCS vent of greater than or equal to 3.58 square inches.

APPLICABILITY:

a. PORVs: In MODES 4, 5 and 6, during cooldown when the temperature of any RCS cold leg is greater than or equal to 161°F and less than or equal to 286°F, and during heatup when the temperature of any RCS cold leg is greater than or equal to 142°F and less than or equal to 295°F.

b. SDCRVs: In MODES 4, 5 and 6, during cooldown when the temperature ~~of~~ any RCS cold leg is less than or equal to 161°F, and during heatup, when the temperature of any RCS cold leg is less than or equal to 142°F.

c. RCS Vent: In MODES 5 and 6, during cooldown when the temperature of any RCS cold leg is less than or equal to 286°F, and during heatup when the temperature of any RCS cold leg is less than or equal to 295°F.

ACTION:

a. With a PORV being used for LTOP inoperable, restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through a greater than or equal to 3.58 square inch vent(s) within the next 8 hours.

b. With both PORVs being used for LTOP inoperable, depressurize and vent the RCS through a greater than or equal to 3.58 square inch vent(s) within the next 8 hours.

c. With a SDCRV being used for LTOP inoperable, restore the inoperable SDCRV to OPERABLE status within 7 days or depressurize and vent the RCS through a greater than or equal to 3.58 square inch vent(s) within the next 8 hours.

LIMITING CONDITION FOR OPERATION

ACTION: Continued:

d. With both SDCRVs being used for LTOP inoperable, depressurize and vent the RCS through a greater than or equal to 3.58 square inch vent(s) within the next 8 hours.

e. In the event either the PORVs, SDCRVs or the RCS vent(s) are used to mitigate a RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs, SDCRVs or vent(s) on the transient and any corrective action necessary to prevent recurrence.

f. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE by:

a. In addition to the requirements of Specification 4.0.5, operating the valve through one complete cycle of full travel at least once per 18 months.

3/4.5.3 ECCS SUBSYSTEMS - T_{avg} LESS THAN 325°F

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. One OPERABLE high-pressure safety injection pump, and

b. An OPERABLE flow path capable of taking suction from the refueling water tank on a Safety Injection Actuation Signal and automatically transferring suction to the containment sump on a Sump Recirculation Actuation Signal.

APPLICABILITY: MODES 3* and 4#.

ACTION:

a. With no ECCS subsystem OPERABLE, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

SURVEILLANCE REQUIREMENTS

4.5.3 The ECCS subsystem shall be demonstrated OPERABLE per the applicable Surveillance Requirements of 4.5.2.

*With pressurizer pressure less than 1750 psia.

#One HPSI shall be rendered inoperable prior to entering MODE 5.

3/4.5.4 REFUELING WATER TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water tank shall be OPERABLE with:

a. A minimum contained borated water volume of 417,100 gallons,

b. A boron concentration of between 1720 and 2100 ppm of boron, and

c. A solution temperature of between 55°F and 100°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the refueling water tank inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.4 The RWT shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the contained borated water volume in the tank, and

2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWT temperature when the outside air temperature is less than 55°F or greater than 100°F.

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps in operation, and maintain DNBR above 1.20 during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation, this specification requires that the plant be in at least HOT STANDBY within 1 hour.

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be OPERABLE.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or shutdown cooling loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops (either shutdown cooling or RCS) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single shutdown cooling loop provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two shutdown cooling loops be OPERABLE.

The operation of one reactor coolant pump or one shutdown cooling pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reductions will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a reactor coolant pump in MODES 4 and 5, with one or more RCS cold leg temperatures less than or equal to 286°F during cooldown and 295°F during heatup are provided to prevent RCS pressure transients, caused by energy additions from the secondary system from exceeding the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients by (1) sizing each PORV to mitigate the pressure transient of an inadvertent safety injection actuation in a water-solid RCS with pressurizer heaters energized, (2) restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 40°F above each of the RCS cold leg temperatures, (3) using SDCRVs to mitigate RCP start transients and the transients caused by inadvertent SIAS actuation and charging water, and (4) rendering one HPSI pump inoperable when the RCS is at low temperatures.

3/4.4.2 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. Each safety valve is designed to relieve 212,182 lbs per hour of saturated steam at the valve setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating shutdown cooling loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the Overpressure Protection System provides a diverse means of protection against RCS overpressurization at low temperatures.

SAFETY VALVES (Continued)

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2750 psia. The combined relief capacity of these valves is sufficient to limit the system pressure to within its Safety Limit of 2750 psia following a complete loss of turbine generator load while operating at RATED THERMAL POWER and assuming no reactor trip until the first Reactor Protective System trip setpoint (Pressurizer Pressure-High) is reached (i.e., no credit is taken for a direct reactor trip on the loss of turbine) and also assuming no operation of the pressurizer power-operated relief valve or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Vessel Code.

3/4.4.3 PRESSURIZER

An OPERABLE pressurizer provides pressure control for the Reactor Coolant System during operations with both forced reactor coolant flow and with natural circulation flow. The minimum water level in the pressurizer assures the pressurizer heaters, which are required to achieve and maintain pressure control, remain covered with water to prevent failure, which could occur if the heaters were energized uncovered. The maximum water level in the pressurizer ensures that this parameter is maintained within the envelope of operation assumed in the safety analysis. The maximum water level also ensures that the RCS is not a hydraulically solid system and that a steam bubble will be provided to accommodate pressure surges during operation. The steam bubble also protects the pressurizer code safety valves against water relief. The requirement to verify that on an Engineered Safety Features Actuation test signal concurrent with a loss of offsite power the pressurizer heaters are automatically shed from the emergency power sources is to ensure that the non-Class 1E heaters do not reduce the reliability of or overload the emergency power source. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability to control Reactor Coolant System pressure and establish and maintain natural circulation.

SPECIFIC ACTIVITY (Continued)

The sample analysis for determining the gross specific activity and $\bar{E}$ can exclude the radioiodines because of the low primary coolant limit of 1 microcurie/gram DOSE EQUIVALENT I-131, and because, if the limit is exceeded, the radioiodine level is to be determined every 4 hours. If the gross specific activity level and radioiodine level in the primary coolant were at their limits, the radioiodine contribution would be approximately 1%. In a release of primary coolant with a typical mixture of radioactivity, the actual radioiodine contribution would probably be about 20%. The exclusion of radionuclides with half-lives less than 10 minutes from these determinations has been made for several reasons. The first consideration is the difficulty in identifying short-lived radionuclides in a sample that requires a significant time to collect, transport, and analyze. The second consideration is the predictable delay time between the postulated release of radioactivity from the primary coolant to its release to the environment and transport to the SITE BOUNDARY, which is relatable to at least 30 minutes decay time. The choice of 10 minutes for the half-life cutoff was made because of the nuclear characteristics of the typical primary coolant radioactivity. The radionuclides in the typical primary coolant have half-lives of less than 4 minutes or half-lives of greater than 14 minutes, which allows a distinction between the radionuclides above and below a half-life of 10 minutes. For these reasons the radionuclides that are excluded from consideration are expected to decay to very low levels before thay could be transported from the primary coolant to the SITE BOUNDARY under any accident condition.

Based upon the above considerations for excluding certain radionuclides from the sample analysis, the allowable time of 2 hours between sample taking and completing the initial analysis is based upon a typical time necessary to perform the sampling, transport the sample, and perform the analysis of about 90 minutes. The gross count should be made in a reproducible geometry of sample and counter having reproducible γ or β self-shielding properties. The counter should be reset to a reproducible efficiency versus energy. It is not necessary to identify specific nuclides.

The determination of the contributors to the $\bar{E}$ result should be based upon those energy peaks identifiable with a 95% confidence level. The radiochemical determination of nuclides should be based on multiple counting of the sample with typical counting basis following sampling of less than 1 hour, about 2 hours, about 1 day, about 1 week, and about 1 month.

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 5.2 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure-temperature curve based on steady state conditions (i.e., no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses at the outer wall of the vessel. These stresses are additive to the pressure induced tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Consequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.

The heatup and cooldown limit curves Figures 3.4-2 and 3.4-3 are composite curves which were prepared by determining the most conservative case, with the reactor vessel beltline or flange juncture limiting for heatup, and vessel beltline limiting for cooldown, for the specified heatup and cooldown rates.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence and copper and nickel content of the material in question, can be predicted using Figure B 3/4.4-1 and the recommendations of Regulatory Guide 1.99, Revision 2 (Draft), "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," and the vendor's flux attenuation factors. The heatup and cooldown limit curves Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of the applicable service period, as well as adjustments for possible errors in the pressure and temperature sensing instruments.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73 and 10 CFR Appendix H, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. The surveillance specimen withdrawal schedule is shown in Table 4.4-5. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the delta RT_{NDT} determined from the surveillance capsule is different from the calculated delta RTNDT for the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown on Figures 3.4-2 and 3.4-3 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50.

The maximum RT_{NDT} for all Reactor Coolant System pressure-retaining materials, with the exception of the reactor pressure vessel, has been determined to be 60°F. The Lowest Service Temperature limit line shown on Figures 3.4-2 and 3.4-3 is based upon this RT_{NDT} since Article NB-2332 (Summer Addenda of 1972) of Section III of the ASME Boiler and Pressure Vessel Code requires the Lowest Service Temperature to be RT_{NDT} + 100°F for piping, pumps, and valves. Below this temperature, the system pressure must be limited to a maximum of 20% of the system's hydrostatic test pressure of 3125 psia.

The limitations imposed on the pressurizer heatup and cooldown rates and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two PORVs, two SDCRVs or an RCS vent opening of greater than 3.58 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 286°F during cooldown and 295°F during heatup. The Low Temperature Overpressure Protection System has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) a safety injection actuation in a water-solid RCS with the pressurizer heaters energized or (2) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 40°F above the RCS cold leg temperatures with the pressurizer solid.

3/4.4.10 REACTOR COOLANT SYSTEM VENTS

Reactor Coolant System vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. The OPERABILITY of at least one Reactor Coolant System vent path from the reactor vessel head and the pressurizer steam space ensures the capability exists to perform this function.

The redundancy design of the Reactor Coolant System vent systems serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply, or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the Reactor Coolant System vent system are consistent with the requirements of Item II.b.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

3/4.4.11 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50.55a (g) (6) (i).

Components of the reactor coolant system were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code 1971 Edition and Addenda through Summer 1973.

3/4.5.1 SAFETY INJECTION TANKS

The OPERABILITY of each of the Reactor Coolant System (RCS) safety injection tanks ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the safety injection tanks. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on safety injection tank volume, boron concentration, and pressure ensure that the assumptions used for safety injection tank injection in the safety analysis are met.

The safety injection tank power-operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these safety injection tank isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limits for operation with a safety injection tank inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occuring concurrent with failure of an additional safety injection tank which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one safety injection tank is not available and prompt action is required to place the reactor in a mode where this capability is not required.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two separate and independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the safety injection tanks is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double-ended break of the largest RCS hot leg pipe downward. In addition, each ECCS subsystem provides long-term core cooling capability in the recirculation mode during the accident recovery period.

ECCS SUBSYSTEMS (Continued)

With the RCS temperature below 325°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The trisodium phosphate dodecahydrate (TSP) stored in dissolving baskets located in the containment basement is provided to minimize the possibility of corrosion cracking of certain metal components during operation of the ECCS following a LOCA. The TSP provided this protection by dissolving in the sump water and causing its final pH to be raised to greater than or equal to 7.0.

The requirement for one high pressure safety injection pump to be rendered inoperable prior to entering MODE 5, although the analysis supports actuation of safety injection in a water solid RCS with pressurizer heaters energized, provides additional administrative assurance that a mass addition pressure transient can be relieved by the operation of a single PORV or SDCRV.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensure that at a minimum, the assumptions used in the accident analyses are met and that subsystem OPERABILITY is maintained. Surveillance Requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses. The requirement to dissolve a representative sample of TSP in a sample of RWT water provides assurance that the stored TSP will dissolve in borated water at the postulated post-LOCA temperatures.

3/4.5.4 REFUELING WATER TANK

The OPERABILITY of the Refueling Water Tank (RWT) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWT minimum volume and boron concentration ensure that (1) sufficient water is available within containment to permit recirculation cooling flow to the core, and (2) the reactor will remain subcritical in the cold condition following mixing of the RWT and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 16

TO FACILITY OPERATING LICENSE NO. NPF-16

FLORIDA POWER & LIGHT COMPANY, ET AL.

ST. LUCIE PLANT, UNIT NO. 2

DOCKET NO. 50-389

1.0 INTRODUCTION

By application dated July 15, 1986, as supplemented by two letters dated September 4, 1986 and one letter dated October 10, 1986, Florida Power and Light Company, the licensee, submitted proposed changes to the St. Lucie Plant, Unit No. 2, Technical Specifications. The proposed changes deal with reactor coolant system pressure/temperature limits and reactor coolant system low temperature overpressure protection.

St. Lucie, Unit 2 presently uses power operated relief valves (PORVs) for overpressure protection during low temperature operating conditions. Pressure/temperature (P/T) limits for plant cooldown and heatup are specified in Figure 3.4-3 of the technical specifications (TS). The licensee's analyses of record assume an inadvertent actuation of a safety injection signal which results in the startup of one high pressure safety injection (HPSI) pump and three charging pumps for a postulated mass addition transient. The analyses assume a maximum secondary to primary temperature differential of 100°F for a postulated inadvertent reactor coolant pump (RCP) start event. The results of these analyses support plant operation in the temperature range within which the Low Temperature Overpressure Protection (LTOP) system is required.

The licensee indicated that the P/T limits currently specified in Figure 3.4-3 of the TS will expire in October 1986. A set of new P/T curves (new Figures 3.4-2 and 3.4-3) was submitted for staff review and approval. The licensee also proposed to modify the LTOP system to use (1) shutdown cooling system (SDCS) relief valves (RVs) for LTOP during heatup when any reactor coolant system (RCS) cold leg temperature is less than 142°F and cooldown when any RCS cold leg temperature is less than

161°F, and (2) the PORV's for LTOP during heatup when any RCS cold leg temperature is between 142°F and 295°F and cooldown when any RCS cold leg temperature is between 161°F and 286°F. The proposed lift setpoints are 350 psia and 470 psia for SDCS RV's and PORV's, respectively. To support the new P/T curves, the new LTOP and their setpoints, the licensee provided the results of analyses for the following cases: (1) the energy addition (RCP start) transient (analysis was performed for a secondary to primary temperature differential of 40°F); (2) the mass addition transient when the RCS cold leg temperature is between 200°F and 295°F (assumes one HPSI pump and three charging pumps adding mass to RCS); and (3) the mass addition transient when the RCS cold leg temperature is less than 200°F (assumes two charging pumps adding mass to the RCS). The licensee proposes to change TS sections 1.16, 3.4.1.3, 3.4.1.4.1, 3.4.9.1, 4.4.9.1.2, Figure 3.4-2, Figure 3.4-3, 3.4.9.3, 3.5.3, 3/4.4.1, 3/4.4.9, and 3/4.5.3 to support the proposed changes to the P/T limits and the LTOP system.

2.0 STAFF EVALUATION - LOW TEMPERATURE OVERPRESSURE PROTECTION

The licensee proposes to add the SDCS RV's as a part of the LTOP system. There are two SDCS RV's, each sized for 2300 gpm at a lift set pressure of 350 psia. Each of the SDCS RV's is located in each of the redundant SDCS trains downstream of the SDCS suction isolation valves inside the containment. Each of the SDCS RV's is sufficient to provide LTOP during low temperature operation when the SDCS RV's are aligned to the RCS. Proposed TS 3.4.9.3 requires the following:

a. Two PORV's, each with a lift pressure setting equal to or less than 470 psia in modes 4, 5 and 6 during cooldown when the temperature of any RCS cold leg is greater than or equal to 161°F and less than or equal to 286°F, and during heatup when the temperature of any RCS cold leg is greater than or equal to 142°F and less than or equal to 295°F.

b. Two SDCS RV's with a lift pressure setting equal to or less than 350 psia in modes 4, 5 and 6 during cooldown when the temperature of any RCS cold leg is less than or equal to 161°F, and during heatup when the temperature of any RCS cold leg is less than or equal to 142°F.

c. The RCS depressurized with a RCS vent of greater than or equal to 3.58 square inches in modes 5 and 6 during cooldown when the temperature of any RCS cold leg is less than or equal to 286°F, and during heatup when the temperature of any RCS cold leg is less than or equal to 295°F.

The results of the analyses provided in the licensee's submittals do not provide adequate bases for the proposed TS. Especially, the assumption of only two charging pumps adding water to the RCS for a postulated mass addition event at a RCS cold leg temperature less than 200°F is nonconservative and deviates from the original licensing basis. The attachment lists the required supporting analyses for all low temperature plant operating conditions considering all possible LTOP system alignments. The results of these analyses should show that the peak transient pressures are within the P/T limits specified in Figures 3.4-2 and 3.4-3 of the TS and that the SDCS design pressure is not exceeded during any mass addition transient.

In response to the staff's request, the licensee in a letter dated October 10, 1986, has committed to provide the results of the above required analyses for staff's review and approval. The staff considers that these analyses are confirmative in nature. In the interim, the staff's approval of the licensee's proposed changes of TS is based on the following:

1. Section 5.4.7.2.3 of the updated FSAR indicates that the SDCS RVs (V-3666 and V-3667) are sized to protect the components and piping from overpressure due to inadvertent starting of the charging pumps, HPSI pumps, and pressurizer heaters. These valves have a set pressure of 335 psig and a capacity of 2300 gpm. When calculating the capacity of valves V-3666 and V-3667, the capacity of each valve was taken to be greater than the combined flow rate of two HPSI

pumps, three charging pumps and the fluid forced out of the pressurizer when the backup heaters are actuated. This is a very conservative sizing since the inadvertent actuation of the HPSI pumps and simultaneous energization of the pressurizer heaters is an unlikely coincidence.

2. Based on the design capacities of the HPSI pumps (approximately 700 gpm runout from each pump) and charging pumps (approximately 44 gpm from each pump) the staff has concluded that sufficient design margin exists for each of the SDCS RVS to mitigate a postulated mass addition transient as discussed above.

3. The results of the licensee's analyses show that a PORV at a lift setting at 470 psia is sufficient for LTOP at RCS cold leg temperatures between 200°F and 295°F.

3.0 Staff Evaluation - RCS Pressure/Temperature Limit Figures

The first surveillance capsule report submitted to the staff by the licensee was Babcock and Wilcox Report BAW-1880, entitled "Analysis of Capsule W-83 Florida Power and Light Company St. Lucie Plant Unit No. 2 Reactor Vessel Material Surveillance Program," September 1985. This report was submitted to the NRC via a letter (L-85-423) from J. W. Williams, Jr., to H. R. Denton, dated November 5, 1985.

Pressure-temperature limits must be calculated in accordance with the requirements of Appendix G, 10 CFR Part 50. Pressure-temperature limits are dependent upon the initial RT_{NDT} for the controlling materials in the beltline and closure flange regions of the reactor vessel and the increase of RT_{NDT} resulting from neutron irradiation damage to the beltline materials. USNRC Standard Review Plan Section 5.3.2, NUREG-0800, Revision 1, July 1981, is used to evaluate the acceptability of pressure-temperature limits for reactor vessels.

The increase in RT_{NDT} resulting from neutron irradiation damage is estimated using the methods documented in Draft Regulatory Guide 1.99, Revision 2, "Radiation Damage to Reactor Vessel Materials." Although this regulatory guide is a draft, its methodology is considered by the staff to be the most up-to-date method for predicting neutron irradiation damage. The licensee uses this version of Regulatory Guide 1.99 to compute the shifts in the reference transition temperature of the controlling metal in the reactor vessel wall. This method of predicting neutron irradiation damage to materials is dependent upon the amount of neutron fluence received by the material and the amounts of copper and nickel in the material. The controlling material in the St. Lucie 2 reactor vessel is base metal. The intermediate shell plate, M605-2, experiences the greatest change with regard to RT_{NDT} through end of license (EOL).

The licensee proposed pressure-temperature limits for 5, 10, 15, 20, 25, 30 and 32 (EOL) effective full power years (EFPY). At each of these EFPY there is an associated neutron fluence (E>1MeV) at the inside surface of the reactor vessel. The neutron fluence at the inside surface of intermediate shell plate M605-2 is pertinent. The neutron fluence (E>1MeV) at each EFPY is in proportion to the neutron fluence (E>1MeV) at EOL or 32 EFPY. Neutron fluence (E>1MeV) at EOL is a calculation involving several uncertainties. Results from the reactor vessel material surveillance program (required by Appendix H, 10 CFR Part 50) assist in the calculation of the projected EOL neutron fluence (E>1MeV).

Appendix H, 10 CFR Part 50, Reactor Vessel Material Surveillance Program Requirements, supplements Appendix G, 10 CFR Part 50, Fracture Toughness Requirements, by requiring a program from which fracture toughness test data are obtained from material specimens exposed in surveillance capsules, which are withdrawn periodically from the reactor vessel. Appendix H refers to ASTM E 185 as an approved standard in the conduct of the program. Table 1 of ASTM E 185-82 is used to determine the number of capsules and capsule withdrawal schedule. To use Table 1, the predicted transition temperature shift at the vessel inside surface must be known. Figure B 3/4.4-1 of the St. Lucie 2 Technical Specifications provides the transition temperature shift as a function of neutron fluence (E>1meV). The staff has three values of neutron fluence at EOL for St. Lucie 2. They are 3.64×10^{19} n/cm^2 as obtained from the capsule report submitted November 1985 (L-85-423), 4.79×10^{19} n/cm^2 as obtained from the submittal regarding pressurized thermal shock (PTS) dated January 23, 1985 (L-86-25) and 5.8×10^{19} n/cm^2 as obtained from the response to the request for additional information regarding pressure-temperature limits (PTL) dated September 4, 1986 (L-86-354). Any of the three values of neutron fluence predicts a value of transition temperature shift of at least 200°F. Thus, Table 1 ASTM E 185-2 specifies five surveillance capsules with the following withdrawal schedule:

Capsule	Schedule in EFPY
1st	1.5
2nd	3
3rd	6
4th	15
5th	EOL

Table 4.4-5 of the present Technical Specifications shows three capsules scheduled for withdrawal, one at 1 EFPY, one at 12 EFPY and one at 24 EFPY. Table 4.4-5 should be changed to reflect the capsule withdrawal schedule presented above. Implementing this change would make the Technical Specifications for St. Lucie 1 and St. Lucie 2 consistent with regard to Appendix H, 10 CFR Part 50.

In the Technical Specifications, only Figures 3.4-2 and 3.4-3 are germane to this evaluation. Pressure-temperature limits for the reactor vessel beyond 4 EFPY should consider information derived from the reactor vessel material surveillance program. The St. Lucie 2 program should have a capsule withdrawn at 3 EFPY. The staff believes it is not pertinent to approve pressure-temperature limits significantly beyond the next scheduled capsule removal. As a practical matter, the capsule should be removed at the refueling outage immediately following 3 EFPY. Following capsule withdrawal, new pressure-temperature limits for a future EFPY applicable through the next scheduled capsule withdrawal should be submitted to the staff. The submittal should be made as soon as possible before the expiration of the currently approved pressure-temperature limits.

Report BAW-1880 contains the analysis of the dosimetry in the most recent withdrawn surveillance capsule, capsule W-83. The calculated peak neutron fluence at the end of license using the results from the capsule W-83 dosimetry is 3.64×10^{19} n/cm^2 (E>1MeV). Since the peak neutron fluence from the capsule W-83 dosimetry is less than the 5.8×10^{19} n/cm^2 (E>1MeV) proposed by the licensee to calculate the reactor vessel pressure-temperature limits, the value of 5.8×10^{19} n/cm^2 (E>1MeV) will estimate conservatively the end of license neutron fluence for the reactor pressure vessel intermediate shell plate M605-2. At 5 EFPY the inside surface neutron fluence is 0.91×10^{19} n/cm^2 (E>1MeV).

The reactor vessel inside radius is 86.22 inches and the outside radius is 94.85 inches which yields a reactor vessel wall thickness of 8.625 inches.

Flaws are postulated on the inside surface and the outside surface of the vessel. Distance is measured from the inner radius of the vessel outward. The flaws on the inside surface and outside surface are referred to by location as 1/4 thickness (1/4t) and 3/4 thickness (3/4t), respectively.

The initial RT_{NDT} for intermediate shell plate M605-2 is 10°F. Margin for the adjusted RT_{NDT} is 2 times the square root of the sum of σ_I^2 and σ_Δ^2. Because 10°F is a measured value, σ_I is zero. The value of σ_Δ is 17°F. The resulting margin is 34°F. The copper and nickel content by weight for intermediate shell plate M605-2 are 0.13% and 0.62%, respectively. Based on the copper and nickel content the base metal chemistry factor is 91.

Recently, Draft Regulatory Guide 1.99, Revision 2, was revised to incorporate comments from the public. No substantive changes in the Regulatory Guide occurred. However, the formula for attenuation of neutron fluence through the vessel wall was revised. The neutron fluence on the inside surface of the vessel is attenuated through the vessel wall by the formula $f(x) = f_o \cdot e^{-0.24x}$. Based on Draft Regulatory Guide 1.99, Revision 2, the values for the adjusted RT_{NDT} computed by the staff at 1/4t (inside surface) and 3/4t (outside surface) for intermediate shell plate M605-2 are 119°F and 95°F, respectively. The corresponding values computed by the licensee are 117°F and 84°F.

The values of adjusted RT_{NDT} computed by the staff and licensee differ because the licensee did not use the method of attenuation described in

Draft Regulatory Guide 1.99, Revision 2. Combustion Engineering (CE), which performed the work for the licensee, used a different methodology which yielded neutron fluences of 0.49×10^{19} n/cm^2 and 0.11×10^{19} n/cm^2 at 1/4t and 3/4t, respectively; whereas the neutron fluences computed by the staff using Draft Regulatory Guide 1.99, Revision 2, are 0.54×10^{19} n/cm^2 and 0.19×10^{19} n/cm^2 at 1/4t and 3/4t, respectively. The values of neutron fluence at 1/4t and 3/4t, particularly 3/4t, as calculated by CE are not conservative compared to the values that result from following the methodology in the draft Regulatory Guide.

The criterion ($K_{IR} \geq 1.5K_{Im} + K_{It}$) from Section III, ASME Code, Article G-2000, Vessels, was used to determine the pressure-temperature limit during an inservice hydrostatic test. The flaw on the inside surface (1/4t) is controlling. The pressure-temperature limits computed by the staff and the licensee are essentially the same for the inservice hydrostatic test. A minimum criticality temperature of 253°F is acceptable. With the exception noted below, the curve for pressure-temperature limit at 5 EFPY for inservice hydrostatic testing is acceptable to the staff.

The criterion ($K_{IR} \geq 2K_{Im} + K_{It}$) from Section III, ASME Code, Article G-2000, Vessels was used to determine the pressure-temperature limit during various heat-up and cooldown rates of the reactor vessel. For heat-up, depending on the heat-up rate and pressure, either the flaw on the inside surface

(1/4t) or the outside surface (3/4t) is controlling. For cooldown, the flaw on the inside surface (1/4t) is controlling.

The cooldown curve for 100°F/hr computed by the licensee conservatively bounds the pressure-temperature curve computed by the staff for the same cooldown rate. Pressure-temperature curves in Figure 3.4-3 for cooldown rates of 0°F/hr (isothermal), 30°F/hr, 50°F/hr and 100°F/hr are acceptable to the staff. The heat-up curve for 50°F/hr computed by the licensee does not bound the pressure-temperature curve computed by the staff for the same heat-up rate. The temperature values differ by as much as 5% at 1400 psig. The difference in computed temperature values between the licensee and the staff are due to (1) the methodology used to attenuate the neutron fluence through the vessel wall and (2) the methodology used to determine the temperature gradient through the vessel wall. A difference in calculated temperature of 5% is not acceptable to the staff. When the entire methodology of Draft Regulatory Guide 1.99, Revision 2, is employed (including the expression for fluence attenuation through the vessel wall), the pressure-temperature curve in Figure 3.4-2 for a heat-up rate of 50°F/hr corresponds more closely to an inner surface neutron fluence of 0.725×10^{19} n/cm² (E>1MeV). A neutron fluence of 0.725×10^{19} n/cm² (E>1MeV) corresponds to 4 EFPY. Thus, the curves in Figures 3.4-2 and 3.4-3 drawn for 5 EFPY are acceptable to the staff for only 4 EFPY. Pressure-temperature limits for heat-up and cooldown during core operations are obtained by adding 40°F to the temperature values in Figures 3.4-2 and 3.4-3. The resulting temperature must be greater than or equal to the minimum critically temperature, 253°F.

Values of 168°F for the lowest service temperature and 80°F for the minimum boltup temperature are determined by considerations in the reactor coolant system other than the reactor pressure vessel. These temperature values are acceptable to the staff. A maximum service pressure of 550 psia is acceptable to the staff.

4.0 CONCLUSION

Based on the above evaluation, the staff concludes that the licensee's proposed changes to TS Sections 1.16, 3.4.1.3, 3.4.1.4.1, 3.4.9.1, 4.4.9.1.2, Figure 3.4-2, Figure 3.4-3, 3.4.9.3, 3.5.3, Bases 3/4.4.1 Bases 3/4.4.9, and Bases 3/4.5.3 as submitted in its letter dated October 10, 1986 are acceptable. Acceptance is based in part on the licensee's commitment to submit confirmatory analyses for the staff's review and approval prior to January 15, 1987. Lastly, the staff recommends the Technical Specifications be changed to reflect the scheduled withdrawal of five surveillance capsules, one each at 1.5, 3,6, 15 and 32 (EOL) effective full power years. By letter dated September 4, 1986, the licensee stated that it is currently evaluating the surveillance capsule withdrawal sequence. The licensee stated that, if the current sequence needs to be revised, a proposed technical specification change will be submitted by December 1, 1986. This schedule is acceptable to the staff.

ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously published a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR §51.22(c)(9). Pursuant to 10 CFR §51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

CONCLUSION

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

Date: October 16, 1986

Principal Contributors: C. Liang, R. Wright, E. Tourigny

Attachment as stated

ATTACHMENT

POSSIBLE LTOP SYSTEM ALIGNMENTS AND THE REQUIRED SUPPORTING ANALYSES

HEATUP OR COOLDOWN	RCS COLD LEG TEMPERATURE	LTOP ALIGNMENTS	REQUIRED SUPPORTING ANALYSES REGARDING MASS ADDITION TRANSIENT
Heatup	≦ 142°F	SDCS in operation (TS 3.4.1.4.1) Two SDCS RVs operable (TS 3.4.9.3.b)	(1) Assumption to include a) one SDCS at a lift setting of 350 psia, and, b) one HPSI pump plus three charging pump in operation following a postulated SIS.
Heatup	142°F-295°F	142°F-200°F SDCS in operation (TS 3.4.1.4.1) PORVs operable (TS 3.4.9.3.a) 200°F-295°F SDCS may be in operation (TS 3.4.1.3) PORV operable (TS 3.4.9.3.a)	(2) Same as (1) above (3) Same as (1) above (when SDCS in operation) (4) Assumptions to include a) one PORV @ a lift setting of 470 psia and b) one HPSI pump plus three charging pump in operation following a postulated SIS. (when no SDCS in operation)
Cooldown	≦ 161°F	SDCS in operation (TS 3.4.1.4.1) Two SDCS RV operable (TS 3.4.9.3.b)	(5) Same as (1) above
Cooldown	161°F-286°F	161°F-200°F SDCS in operation (TS 3.4.1.4.1) PORV operable (TS 3.4.9.3.a) 200°F-286°F SDCS may be in operation (TS 3.4.1.3) PORV operable (TS 3.4.9.3.a)	(6) Same as (1) above (7) Same as (1) above (when SDCS in operation) (8) Same as (4) above (when no SDCS in operation)