ATTACHMENT I to JPN-92-028

PROPOSED TECHNICAL SPECIFICATION CHANGES
POWER UPRATE

(JPTS-91-025)

New York Power Authority

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333
DPR-59

1.0 (cont'd)

C. Cold Condition - Reactor coolant temperature ≤212°F.

D. Hot Standby Condition - Hot Standby condition means operation with coolant temperature >212°F, the Mode Switch in Startup/Hot Standby and reactor pressure <1,040 psig.

E. Immediate - Immediate means that the required action will be initiated as soon as practicable considering the safe operation of the unit and the importance of the required action.

F. Instrumentation

1. Functional Test - A functional test is the manual operation or initiation of a system, subsystem, or component to verify that it functions within design tolerances (e.g., the manual start of a core spray pump to verify that it runs and that it pumps the required volume of water).

2. Instrument Channel Calibration - An instrument channel calibration means the adjustment of an instrument signal output so that it corresponds, within acceptable range, and accuracy, to a known value(s) of the parameter which the instrument monitors. Calibration shall encompass the entire instrument channel including actuation, alarm or trip.

3. Instrument Channel - An instrument channel means an arrangement of a sensor and auxiliary equipment required to generate and transmit to a trip system a single trip signal related to the plant parameter monitored by that instrument channel.

4. Instrument Check - An instrument check is a qualitative determination of acceptable operability by observation of instrument behavior during operation. This determination shall include, where possible, comparison of the instrument with other independent instruments measuring the same variable.

5. Instrument Channel Functional Test - An instrument channel functional test means the injection of a simulated signal into the instrument primary sensor where possible to verify the proper instrument channel response, alarm and/or initiating action.

6. Logic System Function Test - A logic system functional test means a test of relays and contacts of a logic circuit from sensor to activated device to ensure components are operable per design intent. Where practicable, action will go to completion: i.e., pumps

1.0 (cont'd)

opened to perform necessary operational activities.

2. At least one door in each airlock is closed and sealed.

3. All automatic containment isolation valves are operable or de-activated in the isolated position.

4. All blind flanges and manways are closed.

N. Rated Power - Rated power refers to operation at a reactor power of 2,536 MWt. This is also termed 100 percent power and is the maximum power level authorized by the operating license. Rated steam flow, rated coolant flow, rated nuclear system pressure, refer to the values of these parameters when the reactor is at rated power (Reference 1).

O. Reactor Power Operation - Reactor power operation is any operation with the Mode Switch in the Startup/Hot Standby or Run position with the reactor critical and above 1 percent rated thermal power.

P. Reactor Vessel Pressure - Unless otherwise indicated, reactor vessel pressures listed in the Technical Specifications are those measured by the reactor vessel steam space sensor.

Q. Refueling Outage - Refueling outage is the period of time between the shutdown of the unit prior to refueling and the startup of the Plant subsequent to that refueling.

R. Safety Limits - The safety limits are limits within which the reasonable maintenance of the fuel cladding integrity and the reactor coolant system integrity are assured. Violation of such a limit is cause for unit shutdown and review by the Atomic Energy Commission before resumption of unit operation. Operation beyond such a limit may not in itself result in serious consequences but it indicates an operational deficiency subject to regulatory review.

S. Secondary Containment Integrity - Secondary containment integrity means that the reactor building is intact and the following conditions are met:

1. At least one door in each access opening is closed.

2. The Standby Gas Treatment System is operable.

3. All automatic ventilation system isolation valves are operable or secured in the isolated position.

T. Surveillance Frequency - Periodic

Z. Top of Active Fuel

The Top of Active Fuel, corresponding to the top of the enriched fuel column of each fuel bundle, is located 352.5 inches above vessel zero, which is the lowest point in the inside bottom of the reactor vessel. (See General Electric drawing No. 919D690BD.)

AA. Rod Density

Rod density is the number of control rod notches inserted expressed as a fraction of the total number of control rod notches. All rods fully inserted is a condition representing 100 percent rod density.

AB. Purge-Purging

Purge or Purging is the controlled process of discharging air or gas from a confinement in such a manner that replacement air or gas is required to purify the confinement.

AC. Venting

Venting is the controlled process of releasing air or gas from a confinement in such a manner that replacement air or gas is not provided or required.

AD. Core Operating Limits Report (COLR)

This report is the plant-specific document that provides the core operating limits for the current operating cycle. These cycle-specific operating limits shall be determined for each reload cycle in accordance with Specification 6.9.A.4. Plant operation within these operating limits is addressed in individual Technical Specifications.

AE. References

1. General Electric Report NEDC-32016P, "Power Uprate Safety Analysis for the James A. FitzPatrick Nuclear Power Plant," December 1991 (proprietary).

2.1 (cont'd)

2. Reactor Water Low Level Scram Trip Setting

Reactor low water level scram setting shall be $\geq$177 in. above the top of the active fuel (TAF) at normal operating conditions.

3. Turbine Stop Valve Closure Scram Trip Setting

Turbine stop valve scram shall be $\leq$10 percent valve closure from full open when the reactor is at or above 29% of rated power.

4. Turbine Control Valve Fast Closure Scram Trip Setting

Turbine control valve fast closure scram control oil pressure shall be set at $500 < P < 850$ psig.

5. Main Steam Line Isolation Valve Closure Scram Trip Setting

Main steam line isolation valve closure scram shall be $\leq$10 percent valve closure from full open.

6. Main Steam Line Isolation Valve Closure on Low Pressure

When in the run mode main steam line low pressure initiation of main steam line isolation valve closure shall be $\geq$825 psig.

BASES

2.1 FUEL CLADDING INTEGRITY

The abnormal operational transients applicable to operation of the FitzPatrick Unit have been analyzed throughout the spectrum of planned operating conditions up to the thermal power condition of 2,536 MWt. The analyses were based upon plant operation in accordance with the operating map given in the current load line limit analysis. In addition, 2,536 MWt is the licensed maximum power level of FitzPatrick, and this represents the maximum steady-state power which shall not knowingly be exceeded.

The transient analyses performed for each reload are described in Reference 2. Models and model conservatism are also described in this reference. As discussed in Reference 4, the core wide transient analysis for one recirculation pump operation is conservatively bounded by two-loop operation analysis, and the flow-dependent rod block and scram setpoint equations are adjusted for one-pump operation. Reference 1 evaluates the safety significance of uprated power operation at 2,536 MWt. This evaluation is consistent with and demonstrates the acceptability of the transient analyses required by Reference 2.

Fuel cladding integrity is assured by the applicable operating limit MCPR for steady state conditions given in the Core Operating Limits Report (COLR). These operating limit MCPR's are derived from the established fuel cladding integrity Safety Limit, and an analysis of abnormal operational transients. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient.

The most limiting transients have been analyzed to determine which result in the largest reduction in CRITICAL POWER RATIO. The type of transients evaluated were increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest delta MCPR. When added to the Safety Limit, the required operating limit MCPR in the Core Operating Limits Report is obtained.

The evaluation of a given transient begins with the system initial parameters shown in the current reload analysis and Reference 2 that are input to the core dynamic behavior transient computer programs described in Reference 2. The output of these programs along with the initial MCPR form the input for the further analyses of the thermally limited bundle with a single channel transient thermal hydraulic code. The principal result of the evaluation is the reduction in MCPR caused by the transient.

2.1 BASES (Cont'd)

The MCPR operating limits in the COLR are conservatively assumed to exist prior to initiation of the transients.

This choice of using conservative values of controlling parameters and initiating transients at the design power level, produces more pessimistic answers than would result by using expected values of control parameters and analyzing at higher power levels.

Steady-state operation without forced recirculation is not permitted. The analysis to support operation at various power and flow relationships has considered operation with either one or two recirculation pumps.

In summary:

- The abnormal operational transients were analyzed to the licensed maximum power level.
- The licensed maximum power level is 2,536 MWt.
- Analyses of transients employ adequately conservative values of the controlling reactor parameters.
- The analytical procedures now used result in a more logical answer than the alternative method of assuming a higher starting power in conjunction with the expected values for the parameters.

A. Trip Settings

The bases for individual trip settings are discussed in the following paragraphs.

1. Neutron Flux Trip Settings

a. IRM Flux Scram Trip Setting

The IRM system consists of 8 chambers, 4 in each of the reactor protection system logic channels. The IRM is a 5-decade instrument which covers the range of power level between that covered by the SRM and the APRM. The 5 decades are covered by the IRM by means of a range switch and the 5 decades are broken down into 10 ranges, each being one-half of a decade in size. The IRM scram trip setting of 120 divisions is active in each range of the IRM. For example, if the instrument were on Range 1, the scram setting would be a 120 divisions for that range; likewise, if the instrument were on range 5, the scram would be 120 divisions on that range. Thus, as the IRM is ranged up to accommodate the increase in power level, the scram trip setting is also ranged up. The most significant sources of reactivity change during the power increase are due to control rod withdrawal. For insequence control rod withdrawal, the rate of change of power is slow enough due to the physical limitation of withdrawing control rods, that heat flux is in equilibrium with the neutron flux and an IRM scram would result in a reactor shutdown well before any Safety Limit is exceeded.

2.1 BASES (cont'd)

3. Turbine Stop Valve Closure Scram Trip Settings

The turbine stop valve closure scram trip anticipates the pressure, neutron flux and heat flux increase that could result from rapid closure of the turbine stop valves. With a scram trip setting of $\leq$10 percent of valve closure from full open, the resultant increase in surface heat flux is limited such that MCPR remains along the Safety Limit even during the worst case transient that assumes the turbine bypass is closed. This scram is bypassed when reactor power is below 29% of rated, as measured by turbine first stage pressure, consistent with the safety analysis discussed in Reference 1.

4. Turbine Control Valve Fast Closure Scram Trip Setting

This turbine control valve fast closure scram anticipates the pressure, neutron flux, and heat flux increase that could result from fast closure of the turbine control valves due to load rejection exceeding the capability of the turbine bypass. The Reactor Protection System initiates a scram when fast closure of the control valves is initiated by the fast acting solenoid valves. This is achieved by the action of the fast acting solenoid valves in rapidly reducing hydraulic control oil pressure at the main turbine control valve actuator disc dump valves. This loss of pressure is sensed by pressure switches whose contacts form the one-out-of-two-twice logic input to the reactor protection system. This trip setting, a nominally 50 percent greater closure time and a different valve characteristic from that of the turbine stop valve, combine to produce transients very similar and no more severe than for the stop valve. No significant change in MCPR occurs. Relevant transient analyses are discussed in Section 14.5 of the Final Safety Analysis Report and Reference 1. This scram is bypassed when reactor power is below 29 percent of rated, as measured by turbine first stage pressure.

5. Main Steam Line Isolation Valve Closure Scram Trip Setting

The low pressure isolation of the main steam lines at 825 psig was provided to give protection against rapid reactor depressurization and the resulting rapid cooldown of the vessel. Advantage was taken of the scram feature which occurs when the main steam line isolation valves are closed, to provide for reactor shutdown so that high power operation at low reactor pressure does not occur, thus providing protection for the fuel cladding integrity safety limit. Operation of the reactor at pressures lower than 825 psig requires that the Reactor Mode Switch be in the Startup position where protection of the fuel cladding integrity safety limit is provided by the APRM high neutron flux scram and the IRM. Thus, the combination of main stream line low pressure isolation and isolation valve closure scram assures the availability of neutron flux scram protection over the entire range of applicability of the fuel cladding integrity safety limit. In addition, the isolation valve closure scram anticipates the pressure and flux transients which occur during normal or inadvertent isolation valve closure. With the scrams set at $\leq$10 percent valve closure, there is no increase in neutron flux.

6. Main Steam Line Isolation Valve Closure on Low Pressure

The low pressure isolation minimum limit at 825 psig was provided to give protection against fast reactor depressurization and the resulting rapid cooldown of the vessel. Advantage was taken of the scram feature which occurs when the main steam line isolation valves are closed to provide for reactor shutdown so that operation at pressures lower than those specified in the thermal hydraulic safety limit does not occur, although operation at a pressure lower than 825 psig would not necessarily constitute an unsafe condition.

2.1 BASES (Cont'd)

C. References

1. General Electric Report, NEDC-32016P, "Power Uprate Safety Analysis for the James A. FitzPatrick Nuclear Power Plant", December 1991 (proprietary).

2. "General Electric Standard Application for Reactor Fuel", NEDE 24011-P-A (Approved revision number applicable at time that reload fuel analyses are performed).

3. (Deleted)

4. FitzPatrick Nuclear Power Plant Single-Loop Operation, NEDO-24281, August, 1980.

1.2 REACTOR COOLANT SYSTEM

APPLICABILITY:

Applies to limits on reactor coolant system pressure.

OBJECTIVE:

To establish a limit below which the integrity of the Reactor Coolant System is not threatened due to an overpressure condition.

SPECIFICATION:

1. The reactor coolant system pressure shall not exceed 1,325 psig at any time when irradiated fuel is present in the reactor vessel.

2.2 REACTOR COOLANT SYSTEM

APPLICABILITY:

Applies to trip settings of the instruments and devices which are provided to prevent the reactor coolant system safety limits from being exceeded.

OBJECTIVE:

To define the level of the process variables at which automatic protective action is initiated to prevent the safety limits from being exceeded.

SPECIFICATION:

1. The Limiting Safety System setting shall be specified below:

   A. Reactor coolant high pressure scram shall be $\leq$1,080 psig.

   B. Reactor coolant system safety/relief valve nominal settings shall be $\leq$1,145 psig. The allowable setpoint error for each safety/relief valve shall be $\pm$1 percent.

1.2 and 2.2 BASES

The reactor coolant pressure boundary integrity is an important barrier in the prevention of uncontrolled release of fission products. It is essential that the integrity of this boundary be protected by establishing a pressure limit to be observed for all operating conditions and whenever there is irradiated fuel in the reactor vessel.

The pressure safety limit of 1,325 psig as measured by the vessel steam space pressure indicator is equivalent to 1,375 psig at the lowest elevation of the Reactor Coolant System. The 1,375 psig value is derived from the design pressures of the reactor pressure vessel and reactor coolant system piping. The respective design pressures are 1250 psig at 575°F for the reactor vessel, 1148 psig at 568°F for the recirculation suction piping and 1274 psig at 575° for the discharge piping. The pressure safety limit was chosen as the lower of the pressure transients permitted by the applicable design codes: 1965 ASME Boiler and Pressure Vessel Code, Section III for pressure vessel and 1969 ANSI B31.1 Code for the reactor coolant system piping. The ASME Boiler and Pressure Vessel Code permits pressure transients up to 10 percent over design pressure (110% x 1,250 - 1,375 psig) and the ANSI Code permits pressure transients up to 20 percent over the design pressure (120% x 1,150 - 1,380 psig). The safety limit pressure of 1,375 psig is referenced to the lowest elevation of the Reactor Coolant System.

The current reload analysis shows that the main steam isolation valve closure transient, with flux scram, is the most severe event resulting directly in a reactor coolant system pressure increase. The reactor vessel pressure code limit of 1,375 psig, given in FSAR Section 4.2, is above the peak pressure produced by the event above. Thus, the pressure safety limit (1,375 psig) is well above the peak pressure that can result from reasonably expected overpressure transients. (See current reload analysis for the curve produced by this analysis.) Reactor pressure is continuously indicated in the control room during operation.

A safety limit is applied to the Residual Heat Removal System (RHRS) when it is operating in the shutdown cooling mode. When operating in the shutdown cooling mode, the RHRS is included in the reactor coolant system.

The numerical safety/relief valve setpoint shown in 2.2.1.B is justified by analyses described in the General Electric report NEDC-32016P and assures that the structural acceptance criteria set forth in the Mark I Containment Short Term Program are satisfied.

3.1 BASES (cont'd)

is discharged from the reactor by a scram can be accommodated in the discharge piping. Each scram discharge instrument volume accommodates in excess of 34 gallons of water and is the low point in the piping. No credit was taken for this volume in the design of the discharge piping as concerns the amount of water which must be accommodated during a scram.

During normal operation the discharge volume is empty; however, should it fill with water, the water discharged to the piping from the reactor could not be accommodated, which would result in slow scram times or partial control rod insertion. To preclude this occurrence, level detection instruments have been provided in each instrument volume which alarm and scram the reactor when the volume of water reaches 34.5 gallons. As indicated above, there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or amount of insertion of the control rods. This function shuts the reactor down while sufficient volume remains to accommodate the discharged water and precludes the situation in which a scram would be required but not be able to perform its function adequately.

A Source Range Monitor (SRM) System is also provided to supply additional neutron level information during startup but has no scram functions (reference paragraph 7.5.4 FSAR).

The IRM high flux and APRM $\leq$15% power scrams provide adequate coverage in the startup and intermediate range. Thus, the IRM and APRM systems are required to be operable in the refuel and startup/hot standby modes. The APRM $\leq$120% power and flow referenced scrams provide required protection in the power range (reference FSAR Section 7.5.7). The power range is covered only by the APRMs. Thus, the IRM system is not required in the run mode.

The high reactor pressure, high drywell pressure, reactor low water level and scram discharge volume high level scrams are required for startup and run modes of plant operation. They are, therefore, required to be operational for these modes of reactor operation.

The requirement to have the scram functions indicated in Table 3.1-1 operable in the refuel mode assures that shifting to the refuel mode during reactor power operation does not diminish the protection provided by the Reactor Protection System.

Turbine stop valve closure occurs at 10 percent of valve closure. Below 29% of rated reactor power, the scram signal due to turbine stop valve closure is bypassed because the flux and pressure scrams are adequate to protect the reactor.

3.1 BASES (cont'd)

Turbine control valves fast closure initiates a scram based on pressure switches sensing electro-hydraulic control (EHC) system oil pressure. The switches are located between fast closure solenoids and the disc dump valves, and are set relative (500<P<850 psig) to the normal (EHC) oil pressure of 1,600 psig so that based on the small system volume, they can rapidly detect valve closure or loss of hydraulic pressure.

The requirement that the IRM's be inserted in the core when the APRM's read 2.5 indicated on the scale in the start-up and refuel modes assures that there is proper overlap in the neutron monitoring system functions and thus, that adequate coverage is provided for all ranges of reactor operation.

B. The limiting transient which determines the required steady state MCPR limit depends on cycle exposure. The operating limit MCPR values as determined from the transient analysis in the current reload submittal for various core exposures are specified in the Core Operating Limits Report (COLR).

The ECCS performance analyses assumed reactor operation will be limited to MCPR = 1.20, as described in NEDO-21662 and NEDC-31317P including latest revision, errata and addenda. The Technical Specifications limit operation of the reactor to the more conservative MCPR based on consideration of the limiting transient as specified in the COLR.

TABLE 3.1-1 (cont´d)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Minimum No. of Operable Instrument Channels Per Trip System (1)	Trip Function	Trip Level Setting[1]	Modes in Which Function Must be Operable: Refuel (6)	Startup	Run	Total Number of Instrument Channels Provided by Design for Both Trip Systems	Action (1)
2	APRM Downscale	≥ 2.5 indicated on scale (9)			X	6 Instrument Channels	A or B
2	High Reactor Pressure	≤ 1,060 psig	X(8)	X	X	4 Instrument Channels	A
2	High Drywell Pressure	≤ 2.7 psig	X(7)	X(7)	X	4 Instrument Channels	A
2	Reactor Low Water Level	≥ 177 in. above TAF	X	X	X	4 Instrument Channels	A
3	High Water Level in Scram Discharge Volume	≤ 34.5 gallons per Instrument Volume	X(2)	X	X	8 Instrument Channels	A
2	Main Steam Line High Radiation	≤ 3x normal full power background (16)	X	X	X	4 Instrument Channels	A
4	Main Steam Line Isolation Valve closure	≤ 10% valve closure			X(5)	8 Instrument Channels	A
4	Turbine Stop Valve Closure	≤ 10% valve closure			X(4) (5)	8 Instrument Channels	A or C

TABLE 3.1-1 (cont´d)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

NOTES OF TABLE 3.1-1

1. There shall be two operable or tripped trip systems for each function, except as specified in 4.1.D. From and after the time that the minimum number of operable instrument channel for a trip system cannot be met, that affected trip system shall be placed in the safe (tripped) condition, or the appropriate actions listed below shall be taken.

    A. Initiate insertion of operable rods and complete insertion of all operable rods within four hours.

    B. Reduce power level to IRM range and place Mode Switch in the Startup Position within eight hours.

    C. Reduce power to less than 29 percent of rated.

2. Permissible to bypass, if Refuel and Shutdown positions of the Reactor Mode Switch.

3. Deleted.

4. Bypassed when the reactor power is less than 29 percent of rated.

5. The design permits closure of any two lines without a scram being initiated.

6. When the reactor is subcritical and the reactor water temperature is less than 212°F, only the following trip functions need to be operable:

    A. Mode Switch in Shutdown.

    B. Manual Scram.

ATTACHMENT III to JPN-92-028

NEDC-32016P "POWER UPRATE SAFETY ANALYSIS FOR THE JAMES A. FITZPATRICK NUCLEAR POWER PLANT"

(JPTS-91-025)

New York Power Authority

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
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3.5 (Cont'd)

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C. HIGH PRESSURE COOLANT INJECTION (HPCI SYSTEM)

1. The HPCI System shall be operable whenever the reactor pressure is greater than 150 psig and reactor coolant temperature is greater than 212°F and irradiated fuel is in the reactor vessel, except as specified below:

4.5 (Cont'd)

C. HIGH PRESSURE COOLANT INJECTION (HPCI SYSTEM)

Surveillance of HPCI System shall be performed as follows provided a reactor steam supply is available. If steam is not available at the time the surveillance test is scheduled to be performed, the test shall be performed within 10 days of continuous operation from the time steam becomes available.

1. HPCI System testing shall be as specified in 4.5.A.1.a, b, c, d, f, and g except that the HPCI pump shall deliver at least 4,250 gpm against a system head corresponding to a reactor vessel pressure of 1,195 psig to 150 psig.

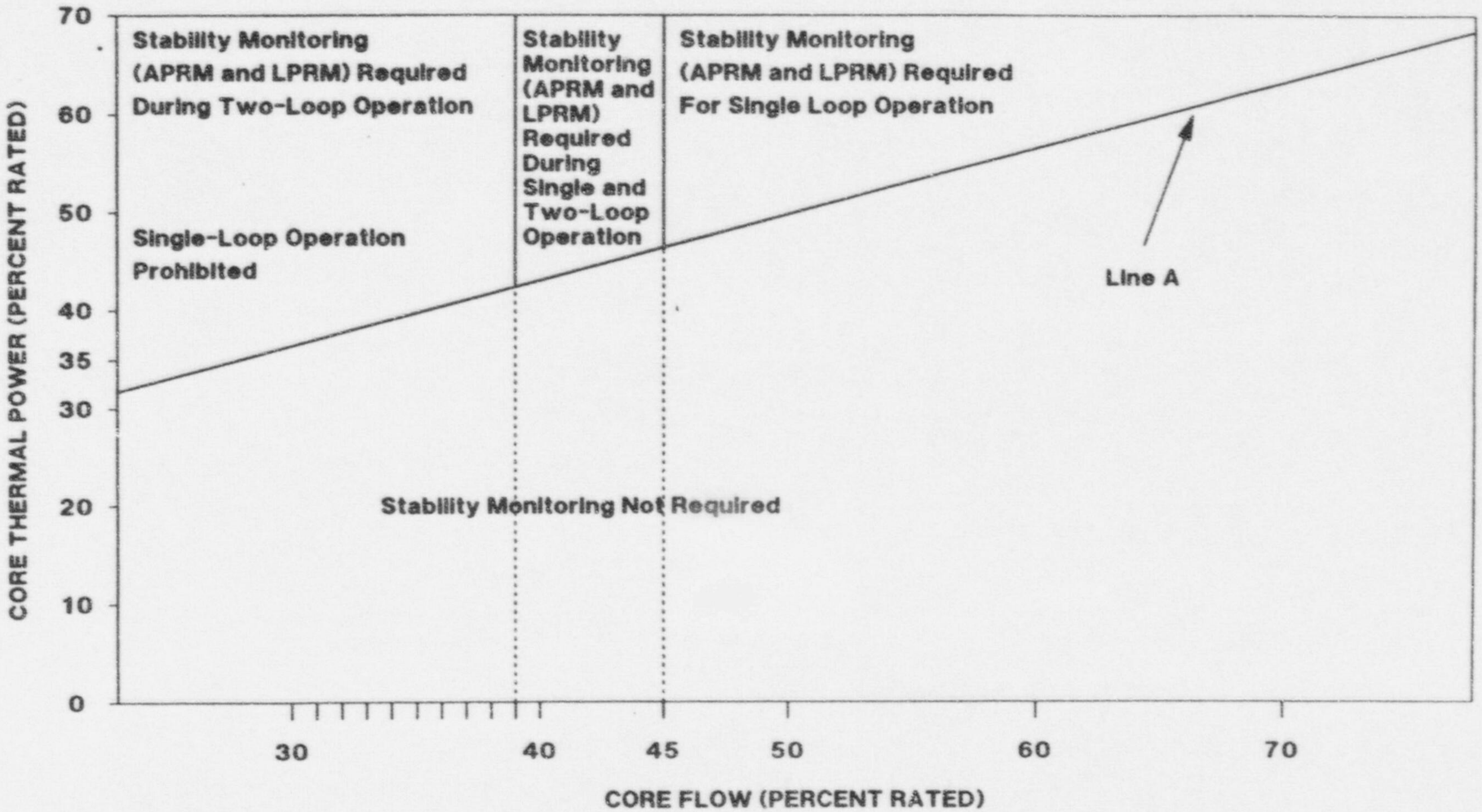
3.5 (cont'd)

4.5 (cont'd)

The RCIC pump shall deliver at least 400 gpm for a system head corresponding to a reactor pressure of 1,195 psig to 150 psig.

2. When it is determined that the RCIC System is inoperable at a time when it is required to be operable, the HPCI System shall be verified to be operable immediately and daily thereafter.

Figure 3.5-1
Thermal Power and Core Flow Limits of
Specifications 3.5.J.1, 3.5.J.2, and 3.5.J.3



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3.6 (cont'd)

B. Deleted

C. Coolant Chemistry

1. The reactor coolant system radioactivity concentration in water shall not exceed the equilibrium value of 0.2 μCi/gm of dose equivalent I-131. This limit may be exceeded, following a power transient, for a maximum of 48 hr. During this iodine activity transient the iodine concentrations shall not exceed the equilibrium limits by more than a factor of 10 whenever the main steamline isolation valves are open. The reactor shall not be operated more than 5 percent of its annual power operation under this exception to the equilibrium limits. If the iodine concentration exceeds the equilibrium limit by more than a factor of 10, the reactor shall be placed in a cold condition within 24 hr.

4.6 (cont'd)

7. Reactor Vessel Flux Monitoring

The reactor vessel Flux Monitoring Surveillance Program complies with the intent of the May, 1983 revision to 10 CFR 50, Appendices G and H. The next flux monitoring surveillance capsule shall be removed after 15 effective full power years (EFPYs) and the test procedures and reporting requirements shall meet the requirements of ASTM E 185-82.

B. Deleted

C. Coolant Chemistry

1. a. A sample of reactor coolant shall be taken at least every 96 hr and analyzed for gross gamma activity.

b. Isotopic analysis of a sample of reactor coolant shall be made at least once/month.

c. A sample of reactor coolant shall be taken prior to startup and at 4 hr intervals during startup and analyzed for gross gamma activity.

d. During plant steady state operation and following an offgas activity increase (at the Steam Jet Air Ejectors) of 10,000 μCi/sec within a 48 hr. period or a power level change of $\geq$20 percent of full rated power/hr reactor coolant samples shall be taken and analyzed for gross gamma activity. At least three samples will be taken at 4 hr intervals. These sampling requirements may be omitted whenever the equilibrium I-131 concentration in the reactor coolant is less than 0.007 μCi/ml.

3.6 and 4.6 BASES (cont'd)

The expected neutron fluence at the reactor vessel wall can be determined at any point during plant life based on the linear relationship between the reactor thermal power output and the corresponding number of neutrons produced. Accordingly, neutron flux wires were removed from the reactor vessel with the surveillance specimens to establish the correlation at the capsule location by experimental methods. The flux distribution at the vessel wall and 1/4 thickness (1/4T) depth was analytically determined as a function of core height and azimuth to establish the peak flux location in the vessel and the lead factor of the surveillance specimens.

Regulatory Guide 1.99, Revision 2 is used to predict the shift in RT_{NDT} as a function of fluence in the reactor vessel beltline region. An evaluation of the irradiated surveillance specimens, which were withdrawn from the reactor in April, 1985 (6 EFPY), shows a shift in RT_{NDT} less than that predicted by Regulatory Guide 1.99, Revision 2.

Operating limits for the reactor vessel pressure and temperature during normal heatup and cooldown, and during in-service hydrostatic and leak testing were established using 10 CFR 50 Appendix G, May, 1983 and Appendix G of the Summer 1984 Addenda to Section III of the ASME Boiler and Pressure Vessel Code. These operating limits assure that the vessel could safely accommodate a postulated surface flaw having a depth of 0.24 inch at the flange-to-vessel junction, and one-quarter of the material thickness at all other reactor vessel locations and discontinuity regions. For the purpose of setting these operating limits, the reference temperature, RT_{NDT}, of the vessel material was estimated from impact test data taken in accordance with the requirements of the Code to which the vessel was designed and manufactured (1965 Edition including Winter 1966 addenda). The RT_{NDT} values for the reactor vessel flange region and for the reactor vessel shell beltline region are 30°F, based on fabrication test reports. The RT_{NDT} for the remainder of the vessel is 40°F.

The first surveillance capsule containing test specimens was withdrawn in April, 1985 after 6 EFPY. The test specimens removed were tested according to ASTM E 185-82 and the results are in GE report MDE-49-0386. The next surveillance capsule will be removed after 15 EFPYs of operation and the results of the examination used as a basis for revision of Figure 3.6-1 curves A, B and C for operation of the plant after 16 EFPYs.

Figure 3.6-1 is comprised of three parts: Part 1, Part 2, and Part 3. Parts 1, 2, and 3 establish the pressure-temperature limits for plant operations through 12, 14, and 16 Effective Full Power Years (EFPY) respectively. The appropriate figure and the pressure-temperature curves are dependent on the number of accumulated EFPY. Figure 3.6-1, Part 1 is for operation through 12 EFPY, Figure 3.6-1, Part 2 is for operation at greater than 12 EFPY through 14 EFPY, and Figure 3.6-1, Part 3 is for operation at greater than 14 EFPY through 16 EFPY. The curves contained in Figure 3.6-1 are developed from the General Electric Report DRF 137-0010, "Implementation of Regulatory Guide 1.99, Revision 2 for the James A. FitzPatrick Nuclear Power Plant," dated June, 1989.

Figure 3.6-1 curve A establishes the minimum temperature for hydrostatic and leak testing required by the ASME Boiler and Pressure Vessel Code, Section XI. Test pressures for in-service hydrostatic and leak testing are a function of the testing temperature and the component material. Accordingly, the maximum hydrostatic test pressure will be 1.1 times the operating pressure or about 1,144 psig.

3.6 and 4.6 BASES (cont'd)

B. Deleted

C. Coolant Chemistry

A radioactivity concentration limit of 20 μCi/ml total iodine can be reached if the gaseous effluents are near the limit as set forth in Radiological Effluent Technical Specification Section 3.2.a if there is a failure or a prolonged shutdown of the cleanup demineralizer.

In the event of a steam line rupture outside the drywell, a more restrictive coolant activity level of 0.2μCi/gm of dose equivalent I-131 was assumed. With this coolant activity level and adverse meteorological conditions, the calculated radiological dose at the site boundary would be less than 30 rem to the thyroid. The reactor water sample will be used to assure that the limit of Specification 3.6.C is not exceeded. The total radioactive iodine activity would not be expected to change rapidly over a period of 96 hr. In addition, the trend of the stack offgas release rate, which is continuously monitored, is a good indicator of the trend of the iodine activity in the reactor coolant. Also during reactor startups and large power changes which could affect iodine levels, samples of reactor coolant shall be analyzed to insure iodine concentrations are below allowable levels. Analysis is required whenever the I-131 concentration is within a factor of 100 of its allowable equilibrium value. The necessity for continued sampling following power and offgas transients will be reviewed within 2 years of initial plant startup.

The surveillance requirements 4.6.C.1 may be satisfied by a continuous monitoring system capable of determining the total iodine concentration in the coolant on a real time basis, and annunciating at appropriate concentration levels such that sampling for isotopic analysis can be initiated. The design details of such a system must be submitted for evaluation and accepted by the Commission prior to its implementation and incorporation in these Technical Specifications.

Since the concentration of radioactivity in the reactor coolant is not continuously measured, coolant sampling would be ineffective as a means to rapidly detect gross fuel element failures. However, some capability to detect gross fuel element failures is inherent in the radiation monitors in the offgas system and on the main steam lines.

Materials in the Reactor Coolant System are primarily 304 stainless steel and Zircaloy fuel cladding. The reactor water chemistry limits are established to prevent damage to these materials. Limits are placed on chloride concentration and conductivity. The most important limit is that placed on chloride concentration to prevent stress corrosion cracking of the stainless steel. The attached graph, Fig. 4.6-1, illustrates the results of tests on stressed 304 stainless steel specimens. Failures occurred at concentrations above the curve; no failures occurred at concentrations below the curve. According to the data, allowable chloride concentrations could be set several orders of magnitude above the established limit, at the oxygen concentration (0.2-0.3 ppm) experienced during power operation. Zircaloy does not exhibit similar stress corrosion failures.

However, there are various conditions under which the dissolved oxygen content of the reactor coolant water could be higher than 0.2-0.3 ppm, such as refueling, reactor startup, and hot standby. During these periods with steaming rates less

4.7 (cont'd)

(4.) See table 4.7-2 for exceptions.

(5.) Acceptance criterion - The combined leakage rate for all penetrations and valves subject to type B and C tests shall be less than 0.60 La. Leakage from containment isolation valves that are sealed with fluid from a seal system may be excluded when determining the combined leakage rate provided that the installed isolation valve seal-water system fluid inventory is sufficient to assure the sealing function for at least 30 days.

d. Other leak rate tests

(1) The leakage rate for containment isolation valves 10-AOV-68A, B (penetration X-13A, B) for Low Pressure Coolant Injection system and 14-AOV-13A, B (penetration X-16A, B) for Core Spray System shall be less than 11 cubic feet per minute per valve (pneumatically tested at 45 psig with ambient temperature) or 10 gallons per minute per valve (hydrostatically) tested at 1,035 psig with ambient temperature.

3.7 BASES

A. Primary Containment

The integrity of the primary containment and operation of the Emergency Core Cooling Systems in combination limit the offsite doses to values less than those specified in 10 CFR 100 in the event of a break in the Reactor Coolant System piping. Thus, containment integrity is required whenever the potential for violation of the Reactor Coolant System integrity exists. Concern about such a violation exists whenever the reactor is critical and above atmospheric pressure. An exception to the requirement to maintain primary containment integrity is allowed during core loading and during low power physics testing when ready access to the reactor vessel is required. There will be no pressure on the system at this time, which will greatly reduce the chances of a pipe break. The reactor may be taken critical during this period, however, restrictive operating procedures and operation of the RWM in accordance with Specification 3.3.B.3 minimize the probability of an accident occurring. Procedures in conjunction with the Rod Worth Minimizer Technical Specifications limit individual control worth such that the drop of any in-sequence control rod would not result in a peak fuel enthalpy greater than 280 calories/gm. In the unlikely event that an excursion did occur, the reactor building and Standby Gas Treatment System, which shall be operational during this time, offers a sufficient barrier to keep offsite doses well within 10 CFR 100.

The pressure suppression pool water provides the heat sink for the Reactor Coolant System energy release following a postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat released during reactor coolant system blowdown from 1,040 psig.

Since all of the gases in the drywell are purged into the pressure suppression chamber air space during a loss of coolant accident, the pressure resulting from isothermal compression plus the vapor pressure of the liquid must not exceed 56 psig, the suppression chamber design pressure. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber (Section 5.2).

3.7 BASES (cont'd)

Using the minimum or maximum downcomer submergence levels given in the specification, containment pressure during the design basis accident is approximately 45 psig which is below the design of 56 psig. The minimum downcomer submergence of 51.5 in. results in an approximate suppression chamber water volume of 105,900 ft.[3] The majority of the Bodega tests (9) were run with a submerged length of 4 ft. and with complete condensation. Thus, with respect to downcomer submergence, this specification is adequate. Additional JAFNPP specific analyses done in connection with the Mark I Containment-Suppression Chamber Integrity Program indicate the adequacy of the specified range of submergence to ensure that dynamic forces associated with pool swell do not result in overstress of the suppression chamber or associated structures. Level instrumentation is provided for operator use to maintain downcomer submergence within the specified range.

The maximum temperature at the end of blowdown tested during the Humboldt Bay (10) and Bodega Bay tests was 170°F, and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperatures above 170°F.

Containment analyses predict a 46°F increase in pool water temperature, after complete LOCA blowdown. These analyses assumed an initial suppression pool water temperature of 95°F and a rated reactor power of 2536 MWt. LOCA analyses in Section 14.6 of the FSAR also assume an initial 95°F pool temperature. Therefore, complete condensation is assured during a LOCA because the maximum pool temperature (141°F) is less than the 170°F temperature seen during the Bodega Bay tests.

For an initial maximum suppression chamber water temperature of 95°F, assuming the worst case complement of containment cooling pumps (one LPCI pump and two RHR service water pumps), containment pressure is required to maintain adequate net positive suction head (NPSH) for the core spray and LPCI pumps.

Limiting suppression pool temperature to 105°F during RCIC, HPCI, or relief valve operation, when decay heat and stored energy are removed from the primary system by discharging reactor steam directly to the suppression chamber assures adequate margin for a potential blowdown any time during RCIC, HPCI, or relief valve operation.

Experiments indicate that unacceptably high dynamic containment loads may result from unstable condensation when suppression pool water temperatures are high near SRV discharges. Action statements limit the maximum pool temperature to assure stable condensation. These actions include: limiting the maximum pool temperature of 95°F during normal operation; initiating a reactor scram if during a transient (such as a stuck open SRV) pool temperature exceeds 110°F; and depressurizing the reactor if pool temperature exceeds 120°F. T-quenchers diffuse steam discharged from SRVs and promote stable condensation. The presence of T-quenchers and compliance with these action statements assure that stable condensation will occur and containment loads will be acceptable.

NEDC-24361P (August 1981) summarizes analyses performed to predict pool temperatures and containment loads during plant transients using these temperature limits at a power level of 2535 MWt (104% of rated). NEDC-24361P also substantiates the acceptability of the plant design using the local pool limits of NUREG-0661. NEDO-30832 (December 1984) shows that SRV condensation loads are low compared to other design loads for plants with T-quenchers. NEDO-30832 describes why local pool temperatures need not be analyzed at a rated power level of 2536 MWt.

4.7 BASES

A. Primary Containment

The water in the suppression chamber is used only for cooling in the event of an accident; i.e., it is not used for normal operation; therefore, a daily check of the temperature and volume is adequate to assure that adequate heat removal capability is present.

The primary containment preoperational test pressures are based upon the calculated primary containment pressure response corresponding to the design basis loss-of-coolant accident. The peak drywell pressure would be about 45 psig which would rapidly reduce to 27 psig within 30 sec. following the pipe break. Following the pipe break, the suppression chamber pressure rises to 26 psig within 30 sec, equalizes with drywell pressure and thereafter rapidly decays with the drywell pressure decay (14).

The design pressure of the drywell and suppression chamber is 56 psig(15). The design basis accident leakage rate is 0.5 percent/day at a pressure of 45 psig. As pointed out above, the drywell and suppression chamber pressure following an accident would equalize fairly rapidly. Based on the primary containment pressure response and the fact that the drywell and suppression chamber function as a unit, the primary containment will be tested as a unit rather than the individual components separately.

Design basis accidents were evaluated as discussed in Section 14.6 of the FSAR and the power uprate safety evaluation, Reference 18. The whole body and thyroid doses in the control room, low population zone (LPZ) and site boundary meet the requirements of 10 CFR Parts 50 and 100. The technical support center (TSC), not designed to these licensing bases, was also analyzed. The whole body and thyroid dose acceptance criteria used for the main control room are met for the TSC when initial access to the TSC and occupancy of certain areas in the TSC is restricted by administrative control. The LOCA dose evaluation, Reference 19, assumed: the primary containment leak rate was 1.5 volume percent per day; source term releases were in accordance with TID-14844; and the standby gas treatment system filter efficiency was 99% for halogens. These doses are also based on the

(A) ROUTINE REPORTS (Continued)

4. CORE OPERATING LIMITS REPORT

a. Core operating limits shall be established prior to startup from eac cycle, or prior to any remaining portion of a reload cycle for the followin(

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

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

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

1. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P, latest approved version and amendments.
2. "James A. FitzPatrick Nuclear Power Plant SAFER/GESTR - LOCA Loss-of-Coolant Accident Analysis," NEDC-31317P, October, 1986 including latest revision, errata and addenda.
3. "Loss-of-Coolant Accident Analysis for James A. FitzPatrick Nuclear Power Plant," NEDO-21662-2, July, 1977 including latest errata and addenda.

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.

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.

7.0 REFERENCES

(1) E. Janssen, "Multi-Rod Burnout at Low Pressure," ASME Paper 62-HT-26, August 1962.

(2) K.M. Backer, "Burnout Conditions for Flow of Boiling Water in Vertical Rod Clusters," AE-74 (Stockholm, Sweden), May 1962.

(3) FSAR Section 11.2.2.

(4) FSAR Section 4.4.3.

(5) I.M. Jacobs, "Reliability of Engineered Safety Features as a Function of Testing Frequency," Nuclear Safety, Vol. 9, No. 4, July-August 1968, pp 310-312.

(6) Benjamin Epstein, Albert Shiff, UCRL-50451, Improving Availability and Readiness of Field Equipment Through Periodic Inspection, July 16, 1968, p. 10, Equation (24), Lawrence Radiation Laboratory.

(7) I.M. Jacobs and P.W. Mariott, APED Guidelines for Determining Safe Test Intervals and Repair Times for Engineered Safeguards - April 1969.

(8) Bodega Bay Preliminary Hazards Report, Appendix 1, Docket 50-205, December 28, 1962.

(9) C.H. Robbins, "Tests of a Full Scale 1/48 Segment of the Humbolt Bay Pressure Suppression Containment," GEAP-3596, November 17, 1960.

(10) "Nuclear Safety Program Annual Progress Report for Period Ending December 31, 1966, ORNL-4071."

(11) Section 5.2 of the FSAR.

(12) TID 20583, "Leakage Characteristics of Steel Containment Vessel and the Analysis of Leakage Rate Determinations."

(13) Technical Safety Guide, "Reactor Containment Leakage Testing and Surveillance Requirements," USAEC, Division of Safety Standards, Revised Draft, December 15, 1966.

(14) Section 14.6 of the FSAR.

(15) ASME Boiler and Pressure Vessel Code, Nuclear Vessels, Section III. Maximum allowable internal pressure is 62 psig.

(16) 10CFR50.54, Appendix J, Reactor Containment Testing Requirements."

(17) 10CFR50, Appendix J, February 13, 1973.

(18) General Electric Report NEDC-32016P, "Power Uprate Safety Analysis for the James A. FitzPatrick Nuclear Power Plant," December 1991 (proprietary).

(19) James A. FitzPatrick Calculation JAF-CALC-RAD-00008, "Radiological Consequences of Design Basis Accidents at James A. FitzPatrick," November 1991.

(20) General Electric Report GE-NE-187-45-1191P, "Containment Systems Evaluation," (proprietary).

ATTACHMENT II to JPN-92-028

SAFETY EVALUATION FOR PROPOSED TECHNICAL SPECIFICATION CHANGES POWER UPRATE (JPTS-91-025)

I. PURPOSE OF THE PROPOSED CHANGES

The purpose of the proposed changes is to revise the Technical Specifications to permit operation of the James A. FitzPatrick Nuclear Power Plant at an uprated power of 2536 MWt. Engineering analyses and evaluations confirm that the plant can be operated at an uprated power. The increase in the rated power from 2436 MWt to 2536 MWt corresponds to a 4.8 percent increase in rated steam flow (Reference 3, Section 1.2). The increase in rated power remains below the plant design power level of 2,550 MWt which was the basis for the original plant safety evaluation, Reference 14.

The Technical Specification changes necessary for power uprate are identified and evaluated in this safety evaluation. The changes to the Technical Specifications were identified from the results and conclusions of References 1 to 6. These include two generic licensing topical reports prepared by General Electric: NEDC-31897P-A "Generic Guidelines for General Electric Boiling Water Reactor Power Uprate," Reference 1, referred to as LTR-1, and; NEDC-31984P "Generic Evaluations of General Electric Boiling Water Reactor Power Uprate," and Supplement 1, Reference 2, referred to as LTR-2. They also include plant specific analyses: General Electric Report NEDC-32016P "Power Uprate Safety Analysis for the James A. FitzPatrick Nuclear Power Plant," Reference 3, referred to as the PUSAR; Stone & Webster Engineering Corporation "Core Power Uprate Engineering Report for James A. FitzPatrick Nuclear Power Plant," Reference 4, referred to as the Engineering Report; General Electric Report NEDC-31317P-1, Revision 1, "James A. FitzPatrick Nuclear Power Plant SAFER/GESTR Loss-of-Coolant Accident Safety Analysis Report," Reference 5, referred to as the ECCS-LOCA Analysis, and; James A. FitzPatrick Calculation JAF-CALC-RAD-00008, "Radiological Consequences of Design Basis Accidents at James A. FitzPatrick," Reference 6, referred to as the Dose Analysis.

This change request is limited to the changes necessary for operation at power uprate conditions. Additional margin associated with the computer models being used for safet analyses have not been used to relax requirements (e.g., ECCS pump flows) except w required to support operation at uprated power. This change request includes no rec approval of plant operations using special features such as increased core flow or th Extended Operating Domain.

Th PUSAR provides a plant specific safety evaluation for power uprate that disc of the evaluations performed for power uprate. The information in the PUSAR is when directly applicable to a change in order to avoid repetition. PUSAR Table peak containment pressure for a LOCA as 41.2 psig. This is lower than the pe pressure of 45 psig now identified in the Technical Specifications. PUSAR Tab the Technical Specifications (i.e., pages 169, 172, 173, 173a, 188, 193 and 19 changed to reflect the lower pressure. These changes are not being reques order to minimize the changes necessary for operation at uprated power. made after the issuance of the PUSAR.

II. DESCRIPTION AND SAFETY IMPLICATION OF THE PROPOSED CH

The Operating License with its attached Technical Specifications repre conditions which the plant must conform with in order to assure publ as the protection of the environment. The Operating License provid the authorizations and limitations for plant operation. The Technica safety limits, limiting safety system settings and limiting control se analyses and evaluations. The Radiological Effluent Technical Specifications contain the provisions for limiting the release of radioactive materials to unrestricted areas during normal operations.

The proposed changes were identified in a systematic review of the Technical Specifications. The necessity for changes was determined using the generic reports LTRs 1 and 2, and the plant specific PUSAR. Where necessary, the supporting documentation such as the engineering report, the LOCA analysis and the dose analysis were used. These documents provide both generic and plant specific evaluations and, where necessary, reanalyses to support James A. FitzPatrick operations at uprated power. The PUSAR is based on the generic format and content for power uprate licensing reports given in LTR-1. It discusses the scope of the engineering and safety evaluations performed for the James A. FitzPatrick power uprate.

The changes affect the operating parameters of the reactor, operational restrictions, setpoints for safety systems, analytical results and test requirements. There are also administrative changes. The changes in each of these categories are summarized as follows:

- Reactor Parameters: The effect on reactor parameters is limited. Higher power is achieved by control rod pattern adjustments to increase reactor thermal power (changes A.2, 3 and 4) in a more uniform (flattened) power distribution to increase steam flow without increasing core recirculation flow. This requires an increased reactor dome pressure (changes A.1 and 5) for adequate turbine inlet pressure.
- Operational Limits: The increased thermal power requires a change B.1) to the limitation on operation in the high power low flow portion of the power/flow map to limit thermal hydraulic instabilities and power oscillations.
- Setpoints: The increased reactor pressure had a direct impact on the high pressure scram setpoint (changes C.4 and 8) and the safety relief valve setpoint (changes C.5 and 6). Additionally, the bypass for the turbine stop valve closure and control valve fast closure scram was changed (changes C.1, 2, 3, 7 and 9) in proportion to the increase in thermal power.
- Analysis Results: Analyses of uprated power transients and accidents required changes to various technical specifications and their bases. Operational parameters and assumptions used in analyses were revised (changes D.1, 2 and 3) to reflect their use as initial conditions. Revised radiological analyses changed dose results (change D.7). The results of the accident analyses required revisions to properly reflect plant capabilities (changes D.4, 5 and 6).
- Testing: A number of changes to testing requirements resulted from power uprate. The increase to reactor pressure had a direct effect on hydrostatic leakage testing pressure (changes E.3 and 4). The test pressure for HPCI and RCIC pumps was revised (changes E.1 and 2) to reflect SRV setpoints assumed in analyses.
- Administrative: Administrative changes (i.e., adding references, revising references and correcting associated errors) were also made (changes F.1, 2, 3, 4, 5, 6 and 7).

No changes to the Radiological Effluent Technical Specifications were identified. For each Technical Specification change, this safety evaluation identifies the specific change proposed, the purpose of the change and the safety implications of the change. This information is presented for each page that is effected so that the need for each change can be clearly identified and its safety significance evaluated. Referencing between the page changes is used to avoid unnecessary repetition of information.

The proposed changes, presented page by page, are as follows:

A. Reactor Parameters

1. Page 2, Specification 1.0.D - Definition of Hot Standby Condition

a. DESCRIPTION

Replace the value "1,005 psig" with the value "1,040 psig."

b. PURPOSE

The change revises the definition of the hot standby condition to reflect the operating pressure of the reactor at uprated power conditions.

c. SAFETY IMPLICATIONS

This change reflects a revision to reactor dome pressure and redefines the hot standby condition which is based, in part, on the reactor dome pressure at rated power. The reactor dome pressure is one of the initial parameters selected for evaluating power uprate. The basis for selecting this parameter and the safety implications are discussed below.

The pressure in the reactor is measured at the reactor dome. An increase in the reactor vessel dome pressure is required to achieve good control characteristics for the turbine control valves at the uprated power condition. Proper pressure regulation is provided if the control valves are $\leq$97% of their wide open position at the uprated power steam flow. This is equivalent to a turbine inlet pressure of 975 psig. Since there is a 55 psig steam line pressure drop at this inlet pressure, the reactor dome pressure must be at least 1,030 psig. A slightly higher reactor dome pressure of 1,040 psig was chosen based on coordination of the reactor heat balance with the turbine capability. This value is used as the basis for defining the plant operating characteristics and performing plant safety evaluations at uprated power.

Section 1.3 of the PUSAR identifies the increase in the reactor dome pressure to 1,040 psig. The safety implications of operating at this increased thermal power are discussed throughout the balance of the PUSAR considering the thermal hydraulic parameters established from the heat balance at this power level. This safety evaluation defines the safety basis for concluding that there are no significant safety impacts for power uprate operation.

d. ASSOCIATED CHANGES

Changes A.5, C.4, C.5, C.6, C.8, E.3 and E.4 relate to this change.

e. REFERENCES

Reference 3, Section 1.3

2. Page 5, Specification 1.0.N - Definition of Rated Power

a. DESCRIPTION

Replace the value "2,436 MWt" with the value "2,536 MWt" and add "(Reference 1)" to the end of the sentence.

b. PURPOSE

The change revises the definition of rated power to reflect the increased thermal power at uprated power conditions and provides a reference to the safety evaluation submitted in support of power uprate which has been added to Technical Specification page 6a as a reference.

c. SAFETY IMPLICATIONS

This change reflects an increase of 4.1% to the rated thermal power for power uprate and redefines the definition of rated power in the Technical Specifications. The revised thermal power level is the basic parameter for all power uprate evaluations.

The increase in thermal power was evaluated using the reactor heat balance to establish thermal hydraulic parameters. The steam flow from the reactor vessel was increased to approximately match the original design flow. The 4.8% increase in steam flow with an increase in dome pressure of 35 psig provides good turbine operating characteristics without any turbine modifications. This power level is achieved with an increase in the power flow map along existing flow control lines.

Section 1.3 of the PUSAR identifies the increase in reactor rated power to 2,536 MWt. The safety implications of operating at this increased thermal power are discussed throughout the balance of the PUSAR considering the thermal hydraulic parameters established from the heat balance at this power level. This safety evaluation defines the safety basis for concluding that there are no significant safety impacts for power uprate operation.

The addition of the reference to the safety evaluation is administrative in nature and can have no safety impact.

d. ASSOCIATED CHANGES

Changes A.3 and A.4 relate to this change.

e. REFERENCES

Reference 3.

3. Page 15, Bases 2.1 - Fuel Cladding Integrity

a. DESCRIPTION

In the first paragraph replace the value "2436 MWt" with the value "2,536 MWt" in two locations.

In the first sentence of the second paragraph, replace the word "given" with the word "described."

At the end of the second paragraph, add the following two sentences: "Reference 1 evaluates the safety significance of uprated power operation at 2,536 MWt. This evaluation is consistent with and demonstrates the acceptability of the transient analyses required by Reference 2."

b. PURPOSE

The changes revise the Bases to reflect the increased rated thermal power at uprated power conditions and the associated supporting references. The first change identifies the new thermal power level and the reference, added on Technical Specification page 20, is the PUSAR.

c. SAFETY IMPLICATIONS

The safety implications are discussed in change A.2.

d. ASSOCIATED CHANGES

Changes A.2, A.4, F.1, F.2 and F.7 relate to this change.

e. REFERENCES

Reference 3, Section 11
"Reference 2" is "General Electric Standard Application for Reactor Fuel", NEDE 24011-P-A (Approved revision number applicable at time that reload fuel analyses are performed).

4. Page 16, Bases 2.1 - Licensed Maximum Power Level

   a. DESCRIPTION

      Replace the value "2436 MWt" with the value "2,536 MWt."

   b. PURPOSE

      The change revises the Bases to reflect the new maximum licensed power level at uprated power conditions.

   c. SAFETY IMPLICATIONS

      The safety implications are discussed in change A.2.

   d. ASSOCIATED CHANGES

      Changes A.2 and A.3 relate to this change.

   e. REFERENCES

      Reference 3

5. Page 187, Bases 3.7 - Suppression Chamber Blowdown

a. DESCRIPTION

In the second paragraph, replace the value "1,020 psig" with the value "1,040 psig."

b. PURPOSE

The change revises the Bases section to reflect the increased operating pressure. This section identifies the suppression chamber water volume function of absorbing the heat released from reactor coolant blowdown. The discussion currently identifies blowdown from 1,020 psig. This value is an editorial error since the intended blowdown is from the rated power pressure of 1,005 psig (i.e., 1,020 psia). This editorial error was in the original issuance. The change indicates that the blowdown is from the pressure of 1,040 psig at uprated power.

c. SAFETY IMPLICATIONS

The safety implications of the uprated power operating pressure are discussed in change A.1. The safety implications of the temperature rise associated with blowdown are discussed in change D.4. There are no safety implications associated with correcting a typographical error.

d. ASSOCIATED CHANGES

Changes A.1, D.4 and D.6 are related to this change.

e. REFERENCES

Reference 3, Section 4.1

B. Operational Limits

1. Page 134, Figure 3.5-1 - Thermal Power and Core Flow Limits

a. DESCRIPTION

Replace the existing Figure 3.5-1 with the revised Figure 3.5-1.

b. PURPOSE

Revise the core thermal power versus core flow operating map for operation at uprated power.

c. SAFETY IMPLICATIONS

Uprated power will shift the line of core thermal power versus core flow which is used to control thermal hydraulic stability. The core thermal power value at which stability monitoring is not required is reduced by 0.96% at each core flow value on Figure 3.5-1. This reduction is in proportion to the ratio of rated power (2436 MWt) to uprated power (2536 MWt). This change assures that the relationship between thermal power and flow represented by "Line A" on Figure 3.5-1 will not change. Therefore, the thermal power cutoff point used, at various flows, to prevent single loop operation or to require stability monitoring for single and two loop operation remains the same. With no change to these values, the margin of safety remains unchanged. The safety implications have been generically evaluated in Section 3.2 of LTR-2, as noted in Section 2.4 of the PUSAR.

d. ASSOCIATED CHANGES

No changes relate to this change.

e. REFERENCES

Reference 2, Section 3.2
Reference 3, Section 2.4

C. Setpoints

1. Page 11, Specification 2.1.A.3 - Turbine Stop Valve Closure Scram Trip Setting

a. DESCRIPTION

Replace the phrase "above 217 psig turbine first stage pressure" with the phrase "the reactor is at or above 29% of rated power."

b. PURPOSE

The change replaces the turbine stop valve closure scram bypass setpoint pressure with a reference reactor power. The actual pressure setpoint for this bypass has varied over time as the turbine first stage pressure associated with the 30% power level has changed. This variation makes a single pressure setpoint inappropriate. The change also makes the setpoint consistent with its power uprate safety basis, reactor power.

c. SAFETY IMPLICATIONS

The setpoint for the scram (i.e., when the turbine stop valve reaches less than or equal to 10% closure from full open) is not changed. The plant transient analyses at uprated power were performed with the setpoint for scram bypass equivalent to 30% of uprated power as discussed in Sections 5 and 9 of the PUSAR. The proposed setpoint for the bypass is conservatively set at less than 29% of the new rated power.

The generic approach to power uprate discusses bypass of this setpoint in Section F.4.2 of LTR-1. The setpoint bypass is chosen to allow operational margin for a scram so that it can be avoided by transferring steam to the turbine bypass system during turbine generator trips at low power. The transient events below the setpoint bypass are non-limiting from a safety viewpoint allowing two options. The first is to keep the setpoint bypass at the current value (this requires adjustment downward to reflect the higher steam flow at uprated power). The second is to maintain the setpoint bypass at the same power level, perform plant specific analysis and readjust instrument setpoints to reflect the increased pressure at that power level.

An analysis was performed for the James A. FitzPatrick plant assuming that the setpoint bypass was at the same power level. The setpoint will be conservatively maintained when reactor power is at or above 29%. This setpoint bypass is sufficiently high to avoid unnecessary scrams and below the analytically required setpoint for the bypass.

The first stage turbine pressure can vary over the life of the plant at the setpoint power level. Calculating the pressure that is equivalent to the setpoint power level avoids revising the Technical Specifications when a variance occurs. The proposed change requires the pressure setpoint for the bypass to be calculated using current methodologies (Reference 7) that assure accurate control of the bypass. This change preserves the current margins of safety because it is conservative with respect to plant analyses, discussed in Sections 5 and 9 of the PUSAR, and provides for more accurate control of the bypass.

d. ASSOCIATED CHANGES

Changes C.2, C.3, C.7, C.9, F.1, F.2 and F.7 relate to this change.

e. REFERENCES

Reference 1, Section F.4.2
Reference 3, Sections 5 and 9
Reference 7

2. Page 19, Bases 2.1.A.3 - Turbine Stop Valve Closure Scram Trip Setting

a. DESCRIPTION

In the last sentence of Section 2.1.A.3, replace the phrase "turbine steam flow is below 30%" with the phrase "reactor power is below 29%."

In the last sentence of Section 2.1.A.3, add the phrase ", consistent with the safety analysis discussed in Reference 1" at the end of the sentence.

b. PURPOSE

The changes revise the Bases to reflect the change proposed to Technical Specifications 2.1.A.3 and 3.1.A (Table 3.1-1, footnote 4) to the value at which the turbine stop valve closure scram is bypassed and provide a reference to the PUSAR to identify a safety discussion of the supporting analyses.

c. SAFETY IMPLICATIONS

The safety implications are discussed in change C.1.

d. ASSOCIATED CHANGES

Changes C.1, C.3, C.7, C.9, F.1, F.2 and F.7 relate to this change.

e. REFERENCES

Reference 1, Section F.4.2
Reference 3, Sections 5 and 9
Reference 7

3. Page 19, Bases 2.1.A.4 - Turbine Control Valve Fast Closure Scram Trip Setting

a. DESCRIPTION

In Section 2.1.A.4, add the phrase "and Reference 1" to the end of the next to last sentence.

In the last sentence of Section 2.1.A.4, replace the phrase "turbine steam flow is below 30 percent" with the phrase "reactor power is below 29 percent."

b. PURPOSE

The changes revise the Bases to reflect the change proposed to Technical Specification 3.1.A (Table 3.1-1, footnote 4) to the value at which the turbine control valve fast closure scram is bypassed and provide a reference to the PUSAR to identify a safety discussion of the supporting analyses.

c. SAFETY IMPLICATIONS

The safety implications discussed in change C.1 are applicable to the turbine control valve fast closure scram trip bypass setting.

d. ASSOCIATED CHANGES

Changes C.1, C.2, C.7, C.9, F.1, F.2 and F.7 relate to this change.

e. REFERENCES

Reference 1, Section F.4.2
Reference 3, Sections 5 and 9
Reference 7

4. Page 27, Specification 2.2.1.A - Reactor Coolant System Limiting Safety System Setting

a. DESCRIPTION

Replace the value "1,045 psig" with the value "1,080 psig."

b. PURPOSE

The reactor high pressure scram setpoint was revised to reflect changes in the plant operating conditions during power uprate. The current reactor high pressure scram limiting safety system setting of 1,045 psig is increased by 35 psig to 1,080 psig to reflect the 35 psig increase in the steam dome during operation.

c. SAFETY IMPLICATIONS

The acceptability of revised setpoint was confirmed by analysis as discussed in Section 5.1.2.1 of the PUSAR. This change in setpoint is within the design envelope and supports power uprate.

d. ASSOCIATED CHANGES

Changes A.1 and C.8 relate to this change.

e. REFERENCES

Reference 3, Section 5.1.2.1

5. Page 27, Specification 2.2.1.B - Reactor Coolant System Limiting Safety System Setting

a. DESCRIPTION

Replace everything after "settings shall be" except the last sentence and replace it with "$\leq$1,145 psig." The specification now reads "Reactor coolant system safety/relief valve nominal settings shall be $\leq$1,145 psig. The allowable setpoint error for each safety/relief valve shall be $\pm$1 percent."

b. PURPOSE

The reactor safety/relief valve (SRV) setpoints are revised to reflect changes in the plant operating conditions during power uprate.

c. SAFETY IMPLICATIONS

The proposed SRV upper bound setpoint of 1,145 psig represents a 35 psig increase over upper bound setpoint proposed at the currently authorized power level (Reference 8). The 35 psig value is consistent with the increase in the dome pressure which will maintain the same simmer margin (i.e., the difference between the valve spring setpoint pressure and normal operating pressure) and be sufficient to maintain overpressure protection. Section 3.2 of the PUSAR discusses an analysis of the limiting pressurization event, MSIV closure with a failure of valve position scram, using a SRV analysis setpoint (i.e., pressure at which the SRVs' open in the analysis) of 1,179 psig. This allows for a 3% margin over the setpoint of 1,145 psig to account for setpoint drift. SRV operation at 1,179 psig was also used in the transient analyses in Section 9.1 of the PUSAR. The ATWS event discussed in Section 9.3.1 used a 1% margin over the 1,145 psig setpoint. This is consistent with the proposed setting.

The power uprate safety analyses included performance improvement features and equipment out of service assumptions as discussed in Section 1.3.2 of the PUSAR. Two safety relief valves out of service and a single upper bound SRV setpoint were two of the improvement features included. The proposed changes include the single upper bound setpoint since this is how the analyses were performed. The proposed change does not include the out of service allowance or change the 1% allowable setpoint drift since the purpose of the proposed change is to make only those changes necessary for efficient uprated power operation. Changes that can be justified because of the additional margin in current analyses are not being requested as part of this amendment request.

The application (Reference 8), to use a single setpoint at 1,110 psig is pending. The evaluation of SRV containment dynamic loads for power uprate using a SRV setpoint of 1,195 psig (i.e., pressure at which the SRVs' open in the analysis) is provided in Section 4.1.2.2 of the PUSAR. This evaluation is based on the analysis described in Reference 9. Reference 9 was developed in support of the pending Technical Specification change.

d. ASSOCIATED CHANGES

Changes A.1, C.6, E.1, E.2, F.1, F.2 and F.7 relate to this change.

e. REFERENCES

Reference 3, Sections 3.2, 9.1 and 9.3.1
Reference 8
Reference 9

6. Page 29, Bases 1.2 and 2.2 - SRV Settings

a. DESCRIPTION

Delete the first part of the last sentence that reads "The numerical distribution of safety/relief valve setpoints shown in 2.2.1.B (2 @ 1090 psi, 2 @ 1105 psi, 7 @ 1140 psi) is justified by analyses described in the General Electric report NEDO- 24129-1, Supplement 1," and replace it with "The numerical safety/relief valve setpoint shown in 2.2.1.B is justified by analyses described in the General Electric report NEDC-32016P."

b. PURPOSE

This change corrects the Bases by providing the reference used to justify the setpoint change for power uprate.

c. SAFETY IMPLICATIONS

The safety implications of this change are discussed in change C.5.

d. ASSOCIATED CHANGES

Changes A.1, C.5, E.1, E.2, F.1, F.2 and F.7 relate to this change.

e. REFERENCES

Reference 3

7. Page 34, Bases 3.1 - Turbine Stop Valve Closure Scram Trip Setting

a. DESCRIPTION

Replace "217 psig turbine first stage pressure (30 percent of rated)" with "29% of rated reactor power" in the second sentence of the last paragraph.

b. PURPOSE

The change revises the Bases to reflect the change to Table 3.1-1 of Specification 3.1.A.

c. SAFETY IMPLICATIONS

The safety implications are discussed in change C.1.

d. ASSOCIATED CHANGES

Changes C.1, C.2, C.3, C.9, F.1, F.2 and F.7 relate to this change.

e. REFERENCES

Reference 1, Section F.4.2
Reference 3, Sections 5 and 9
Reference 7

8. Page 41a, Table 3.1-1 - Reactor Protection System (SCRAM) Instrumentation Requirement

a. DESCRIPTION

In the trip level setting column, for the high pressure trip function replace the value "$\leq$1045 psig" with the value "$\leq$1,080 psig."

b. PURPOSE

The change revises the reactor high pressure scram setpoint to reflect the reactor operating pressure. The reactor high pressure scram limiting safety system setting of 1,045 psig is increased by 35 psig to 1,080 psig to reflect the 35 psig increase in the steam dome during operation.

c. SAFETY IMPLICATIONS

The safety implications are discussed in change C.4.

d. ASSOCIATED CHANGES

Changes A.1 and C.4 relate to this change.

e. REFERENCES

Reference 3, Section 5.1.2.1

9. Page 42, Table 3.1-1 - Turbine Stop Valve Closure and Turbine Control Valve Fast Closure Scram Trip Setting

a. DESCRIPTION

In note 1.C, replace the value "30 percent" with the value "29 percent."

In note 4, delete the phrase "turbine first stage pressure is less than 217 psig or less than 30 percent" and replace it with the phrase "the reactor power is less than 29 percent."

b. PURPOSE

The change provides the revised value at which the turbine stop and turbine control valves can have their valve closure/fast valve closure scrams bypassed. This also establishes the limit on operating power when instrumentation is not available.

c. SAFETY IMPLICATIONS

The safety implications are discussed in change C.1.

d. ASSOCIATED CHANGES

Changes C.1, C.2, C.3, C.7, F.1, F.2 and F.7 relate to this change.

e. REFERENCES

Reference 1, Section F.4.2
Reference 3, Sections 5 and 9
Reference 7

D. Analysis Results

1. Page 139, Specification 3.6.C.1 - Coolant Chemistry

a. DESCRIPTION

Replace the value "3.1 μCi/gm" with the value "0.2 μCi/gm."

b. PURPOSE

The change reduces the reactor coolant system radioactivity operating limit to be consistent with new accident and transient analyses performed at power uprate conditions.

c. SAFETY IMPLICATIONS

The analyses for James A. FitzPatrick power uprate conditions included an evaluation of transients and accidents as discussed in Section 9 of the PUSAR. These analyses assumed a value of 0.2 microcuries per gram of dose equivalent I-131. The dose analyses for the accident conditions are discussed in Section 9.2 and summarized in Table 9-3 of the PUSAR. The limitations on specific activity in the primary coolant system are the basis for evaluating thyroid and whole body doses from the main steamline failure outside containment.

The value selected for analysis is consistent with NUREG-0123, Revision 3, the BWR Standard Technical Specifications, Reference 10. Section 3/4.4.5 of NUREG-0123 identifies 0.2 microcuries per gram as an interim limit selected by the NRC based upon a parametric evaluation of typical site locations.

These analyses demonstrate the acceptability of operation at the 0.2 microcuries per gram level. This change is more restrictive because it reduces the absolute value of the source term for the main steam line rupture.

d. ASSOCIATED CHANGES

Change D.2 relates to this change.

e. REFERENCES

Reference 3, Section 9.2
Reference 6
Reference 10

2. Page 149, Bases 3.6.C and 4.6.C - Coolant Chemistry And Dose Analysis

a. DESCRIPTION

In the second paragraph, replace the first sentence that says:

> "In the event of a steam line rupture outside the drywell, with this coolant activity level, the resultant radiological dose at the site boundary would be 33 rem to the thyroid, under adverse meteorological conditions assuming no more than 3.1μCi/gm of dose equivalent I-131."

with

> "In the event of a steam line rupture outside the drywell, a more restrictive coolant activity level of 0.2μCi/gm of dose equivalent I-131 was assumed. With this coolant activity level and adverse meteorological conditions, the calculated radiological dose at the site boundary would be less than 30 rem to the thyroid."

b. PURPOSE

The change revises the Bases to make it consistent with the revised dose analysis for main steam line break. This analysis was performed using a revised reactor coolant specific activity. The revised specific activity is now smaller than the specific activity allowed by the Radiological Effluent Technical Specification limit. The change makes this clear.

c. SAFETY IMPLICATIONS

The safety implications are discussed in change D.1.

d. ASSOCIATED CHANGES

Changes D.1, D.7, F.1, F.2 and F.7 are related to this change.

e. REFERENCES

Reference 3, Section 9.2
Reference 6
Reference 10

3. Page 188, Bases 3.7 - Torus Water Volume

a. DESCRIPTION

In the first paragraph, replace the phrase "a minimum suppression chamber" with the phrase "an approximate suppression chamber" and replace the value "105,600 ft^3" with the value "105,900 ft^3."

b. PURPOSE

This change revises the discussion in the Bases of the suppression chamber water volume to reflect the uprated power containment analyses.

c. SAFETY IMPLICATIONS

The revised suppression chamber water volume reflects the latest calculation, Reference 11, of the suppression chamber water volume at minimum water level. The calculated value is 105,930 ft^3. This change represents a minor increase, about 0.3%, in the value used in the original plant calculations. There is no actual change to the water level or volume in the torus. The change is the result of a more accurate calculation. The description of the volume, as approximate, reflects the potential for slight changes in volume with recalculation. The change in calculated volume has no safety impact.

d. ASSOCIATED CHANGES

There are no changes related to this change.

e. REFERENCES

Referance 3, Section 4. i
Heference 11

4. Page 188, Bases 3.7 - Suppression Chamber Water Temperature

a. DESCRIPTION

Replace the third paragraph that says:

> "Using a 40°F rise (Section 5.2 FSAR) in the suppression chamber water temperature and a maximum initial temperature of 95°F, a temperature of 145°F is achieved, which is well below the 170°F temperature which is used for complete condensation."

with:

> "Containment analyses predict a 46°F increase in pool water temperature, after complete LOCA blowdown. These analyses assumed an initial suppression pool water temperature of 95°F and a rated reactor power of 2536 MWt. LOCA analyses in Section 14.6 of the FSAR also assume an initial 95°F pool temperature. Therefore, complete condensation is assured during a LOCA because the maximum pool temperature (141°F) is less than the 170°F temperature seen during the Bodega Bay tests."

b. PURPOSE

This change revises the discussion in the Bases of the calculated temperature rise in the suppression chamber based upon the uprated power analyses and suppression pool temperature Limiting Conditions for Operation.

c. SAFETY IMPLICATIONS

The peak calculated suppression chamber water temperature due to uprated power will increase due to the increased heat in the core. This change does not affect the current Technical Specifications but the Bases are being revised to reflect the current calculations and clearly identify the supporting documentation for the Technical Specifications.

The Bases are clarified to indicate the correct initial temperature assumed in plant LOCA analyses. The existing Bases identifies a 40°F increase in water temperature and a peak blowdown water temperature of 145°F. By inference, the initial water temperature for the LOCA analysis was 105°F based on the allowable 10°F rise for testing in the Technical Specifications. The LOCA analysis assumed that the plant was at the normal operating water temperature limit of 95°F as noted in FSAR Figure 14.6-7. At uprated power, Section 4.1.1.4 of the PUSAR identifies a post blowdown torus water temperature of 141°F when the initial water temperature is 95°F. The rewritten Bases clarifies that the initial water temperature for LOCA analysis was 95°F. There is no safety significance to this change since the LOCA analyses assumed this initial temperature.

The Bases are revised to reflect the increase in torus water temperature due to the LOCA blowdown at uprated power conditions. Section 4.1.1.4 of the PUSAR identifies a 46°F increase in water temperature from LOCA blowdown. The Bases are revised to indicate that there is a 46°F water temperature rise from 95°F to 141°F due to LOCA blowdown. There is no safety significance to this change since the torus water temperature remains well below the 170°F temperature limit for complete condensation based on the Humboldt Bay and Bodega Bay tests. The calculated torus temperatures are within current design values.

d. ASSOCIATED CHANGES

Changes A.5, D.5 and D.6 are related to this change.

e. REFERENCES

Reference 3, Section 4.1

5. Page 188, Bases 3.7 - ECCS Pump NPSH

a. DESCRIPTION

Replace the fourth paragraph which says:

> "For an initial maximum suppression chamber water temperature of 95°F and assuming the normal complement of containment cooling pumps (two LPCI pumps and two RHR service water pumps) containment pressure is not required to maintain adequate net positive suction head (HPSH) for the core spray LPCI and HPCI pumps."

with:

> "For an initial maximum suppression chamber water temperature of 95°F, assuming the worst case complement of containment cooling pumps (one LPCI pump and two RHR service water pumps), containment pressure is required to maintain adequate net positive suction head (NPSH) for the core spray and LPCI pumps."

b. PURPOSE

This change revises the discussion in the Bases of the NPSH requirements for the ECCS pumps. The design basis and analyses do not consider two trains to be available and the Bases should reflect the design condition.

c. SAFETY IMPLICATIONS

The existing Bases section discusses NPSH capabilities of the ECCS pumps for a case where there is no single failure assumed. The change is necessary because no reanalysis of suppression chamber water temperature was performed for the case where two loops of containment cooling were available.

The design basis assumes a single failure and the Bases section has been rewritten to reflect this. The worst case is the failure of an Emergency Diesel Generator (EDG) system which results in the loss of 2 RHRSW pumps and 2 RHR pumps. Two RHRSW pumps and 1 RHR pump (the other is assumed to be discharging to a broken recirculation loop) remain. The suppression chamber rises to 208.7°F, as indicated in Table 4-1 of the PUSAR, when the single failure of one EDG is assumed. The pumps require up to 2 psig of torus pressure at this temperature as discussed in Section 4.1 of the PUSAR and Section 3.9 of the Engineering Report. The use of the containment pressure is consistent with the current plant design bases discussed in Section 6.5.1 of the FSAR (Reference 13). The change does not represent a change in the ability of the ECCS to perform its intended function.

d. ASSOCIATED CHANGES

Changes A.5, D.4 and D.6 are related to this change.

e. REFERENCES

Reference 3, Section 4.1
Reference 4, Section 3.9
Reference 12, Section 6.5.1

6. Page 188, Bases 3.7 - Torus Temperature Limits

a. DESCRIPTION

Delete the last paragraph on the page "Experimental data indicates that excessive steam condensing loads can ... to avoid the regime of potentially high suppression chamber loadings." Replace the deleted material with:

> "Experiments indicate that unacceptably high dynamic containment loads may result from unstable condensation when suppression pool water temperatures are high near SRV discharges. Action statements limit the maximum pool temperature to assure stable condensation. These actions include: limiting the maximum pool temperature of 95°F during normal operation; initiating a reactor scram if during a transient (such as a stuck open SRV) pool temperature exceeds 110°F; and depressurizing the reactor if pool temperature exceeds 120°F. T-quenchers diffuse steam discharged from SRVs and promote stable condensation. The presence of T-quenchers and compliance with these action statements assure that stable condensation will occur and containment loads will be acceptable.
>
> NEDC-24361P (August 1981) summarizes analyses performed to predict pool temperatures and containment loads during plant transients using these temperature limits at a power level of 2535 MWt (104% of rated). NEDC-24361P also substantiates the acceptability of the plant design using the local pool limits of NUREG-0661. NEDO-30832 (December 1984) shows that SRV condensation loads are low compared to other design loads for plants with T-quenchers. NEDO-30832 describes why local pool temperatures need not be analyzed at a rated power level of 2536 MWt."

b. PURPOSE

The change eliminates the discussion in the Bases of the peak suppression pool temperature limit of 160°F used to avoid excessive loads due to steam condensing during blowdown to the torus. This temperature limit was adopted by utilities in 1974 when the phenomenon was initially defined. The power uprate evaluation has identified plant specific and generic analyses that supersede this temperature limit and establish new justifications for the torus temperature limits. The purpose of the change is to reconcile these assessments and their relation to the torus temperature limits.

c. SAFETY IMPLICATIONS

There are no safety concerns associated with this change. The existing paragraph is no longer applicable. The paragraph being deleted was added in Amendment 16 as the basis for the torus temperature limits (i.e., 95°F normal operating, 110°F scram requirement, 120°F isolated reactor depressurization) that were also added.

The NRC identified a local temperature limit (i.e., 200°F) to replace the bulk torus temperature. This limit and its supporting bases were issued in NUREG-0661. Transient analyses performed for FitzPatrick (NEDC-24361P) demonstrate compliance with the acceptance criteria of NUREG-0661. These analyses assumed that the initial temperature of the torus water was 95°F, that reactor scram would be initiated at a water temperature of 110°F and that depressurization would begin at a water temperature of 120°F, if the reactor were isolated.

Subsequent research and testing lead to a conclusion that local temperature limits were not required for plants with quenchers. The bases for this conclusion is provided in NEDO-30832 which demonstrates that the condensation loads with quenchers over the full range of pool temperature up to saturation are low compared to loads due to SRV discharge line air clearing and LOCAs which have already been considered in containment design evaluations.

The conclusion that condensation loads do not require local temperature limits did not eliminate the need to limit the pool water temperature. NEDO-30832 has not eliminated the current licensing basis, NEDC-24361P, but has been relied upon to eliminate the need for further analysis of local pool temperature at uprated power conditions. Limits on water temperature assure that torus water temperature is maintained below the saturation temperature limits in NEDO-30832. Also, pool temperature limits have been used as initial conditions in transient and accident analyses that are part of the design basis for structures and equipment.

d. ASSOCIATED CHANGES

Changes A.5, D.4, D.5 and F.7 are related to this change.

e. REFERENCES

Reference 3, Section 4.1
Reference 15
Reference 16

7. Page 193, Bases 4.7.A - LOCA Dose Analysis

a. DESCRIPTION

Delete the first part of the fourth paragraph "The design basis loss-of-coolant accident ... unlikely event of a design basis loss-of-coolant accident." Replace the deleted material with:

> "Design basis accidents were evaluated as discussed in Section 14.6 of the FSAR and the power uprate safety evaluation, Reference 18. The whole body and thyroid doses in the control room, low population zone (LPZ) and site boundary meet the requirements of 10 CFR Parts 50 and 100. The technical support center (TSC), not designed to these licensing bases, was also analyzed. The whole body and thyroid dose acceptance criteria used for the main control room are met for the TSC when initial access to the TSC and occupancy of certain areas in the TSC is restricted by administrative control. The LOCA dose evaluation, Reference 19, assumed: the primary containment leak rate was 1.5 volume percent per day; source term releases were in accordance with TID-14844; and the standby gas treatment system filter efficiency was 99% for halogens."

b. PURPOSE

The change eliminates the discussion in the Bases of the specific dose results from the LOCA dose calculation. A change was necessary because the dose analysis at uprated power changed the calculational models and results. A reference to the uprate power safety evaluation was also necessary. The change discusses the assumptions used in the dose calculations in the same level of detail for consistency. The results of the dose calculations are discussed generally since it is necessary to show compliance with the acceptance criteria and not calculational results. The results of the dose analysis will be included in an FSAR update following power uprate approval.

c. SAFETY IMPLICATIONS

The dose analyses for power uprate were performed as described above as well as with additional methodologies and assumptions consistent with current NRC acceptance criteria in NUREG-0800, Revision 1. The analyses and specific dose results are discussed in Section 9.2 of the PUSAR. As noted in that section, access to the TSC is administratively restricted based upon measured activity. This control, applicable to all accidents, was initiated as a result of the MSLB dose analysis. The MSLB will result in an unacceptable thyroid dose in the TSC if it is activated immediately after the accident. Using more realistic activity levels (e.g., the design activity level of 0.11 μCi/gm of dose equivalent I-131 rather than the Technical Specification limit of 0.2 μCi/gm) would allow immediate access because the dose following a MSLB at the design activity level would meet 10 CFR 50, Appendix A, GDC 19 dose criteria. This approach assures compliance with federal guidelines and allows access in a reasonable time.

d. ASSOCIATED CHANGES

Changes D.2 and F.7 are related to this change.

e. REFERENCES

Reference 3, Section 9.2
Reference 6

E. Testing

1. <u>Page 117, Specification 4.5.C.1 - HPCI System Surveillance Test Pressure</u>

   a. DESCRIPTION

   Replace the value "1,120 psig" with the value "1,195 psig."

   b. PURPOSE

   The change revises the HPCI test pressure to reflect the analyzed value at which the SRV could be set.

   c. SAFETY IMPLICATIONS

   The HPCI must be able to deliver water to the primary system at the highest pressure allowed by the SRV. The peak pressure the primary system can attain corresponds to the setpoint and drift allowed before the SRV's act to depressurize the system. The change to Specification 2.2.1.A discusses new setpoints for the SRV at 1,145 psig with a 1% setpoint error allowed. This setpoint is consistent with power uprate analyses referred to in Sections 3.2 and 9.1 of the PUSAR which assumed that the SRV would operate at 1,179 psig. However, other analyses have been performed assuming that the SRV operate at 1,195 psig, see Section 4.1.2.2 of the PUSAR. The revised test pressure for the HPCI pump is conservatively based on the highest analyzed pressure. This provides a margin of over 38 psig between the required delivery pressure and the test pressure. The proposed change is therefore conservative.

   d. ASSOCIATED CHANGES

   Changes C.5, C.6 and E.2 relate to this change.

   e. REFERENCES

   Reference 3, Sections 3.2, 4.1.2.2 and 9.1

2. Page 121a, Specification 4.5.E.1 - RCIC System Surveillance Test Pressure

a. DESCRIPTION

Replace the value "1,120 psig" with the value "1,195 psig."

b. PURPOSE

Revise the RCIC test pressure to reflect the analyzed value at which the SRV could be set.

c. SAFETY IMPLICATIONS

The safety implications are discussed in change E.1.

d. ASSOCIATED CHANGES

Changes C.5, C.6 and E.1 relate to this change.

e. REFERENCES

Reference 3, Sections 3.2, 4.1.2.2 and 9.1

3. Page 147, Bases 3.6 and 4.6 - Maximum Hydrostatic Test Pressure

a. DESCRIPTION

In the last sentence on the page, replace the value "1105 psig" with the value "1,144 psig."

b. PURPOSE

The change increases the peak hydrostatic test pressure to reflect the increased reactor operating pressure. The code allows hydrostatic testing to 1.1 times the operating pressure. The power uprate increase in the operating pressure by 35 psig to 1,040 psig results in a 39 psig increase in the peak allowable test pressure to 1,144 psig.

c. SAFETY IMPLICATIONS

There are no safety implications because the new test pressure remains below the FitzPatrick reactor vessel and pressure boundary design pressure of 1250 psig and is significantly below the ASME code allowable peak pressure of 1375 psig. This value remains within the design of the system resulting, as discussed in Sections 3.1 and 3.2 of the PUSAR, in no safety concerns. The safety implications of the increased operating pressure are described in change A.1.

d. ASSOCIATED CHANGES

Changes A.1 and E.4 relate to this change.

e. REFERENCES

Reference 3, Sections 3.1 and 3.2

4. Page 172, Specification 4.7.A.2.d.(1) - Containment Leakage Test Pressure

a. DESCRIPTION

Replace the value "1000 psig" with the value "1,035 psig."

b. PURPOSE

The change revises the leakage testing criteria to reflect the new operating pressure.

c. SAFETY IMPLICATIONS

The are no safety implications associated with this change. The revised pressure is based on the revised system operating pressure discussed in change A.1. This pressure is within the system design limit and is less than the pressure used for hydrostatic and leak rate tests discussed in change E.3. The ASME code requires the hydrostatic test pressure to be less than the normal operating pressure for self seating valves. The differential between operating pressure and test pressure has been maintained. See changes A.1 and E.3 for a further discussion of safety implications.

d. ASSOCIATED CHANGES

Changes A.1 and E.3 are related to this change.

e. REFERENCES

Reference 3

F. Administrative

1. Page 6a, Specification AE - References

a. DESCRIPTION

Add a new specification that reads as follows:

"AE. References

1. General Electric Report NEDC-32016P, "Power Uprate Safety Analysis for the James A. FitzPatrick Nuclear Power Plant," December 1991 (proprietary)."

b. PURPOSE

The definition was added to provide a reference to the PUSAR that was prepared in support of the power uprate application.

c. SAFETY IMPLICATIONS

There are no safety implications associated with the addition of a reference.

d. ASSOCIATED CHANGES

Changes A.3, C.1, C.2, C.3, C.6, C.7, C.9, D.2, D.6, F.2 and F.7 are related to this change.

e. REFERENCES

Reference 3

2. Page 20, Bases 2.1.C - References

a. DESCRIPTION

In item 1, replace the word "(Deleted)" and insert the reference "General Electric Report, NEDC-32016P, "Power Uprate Safety Analysis for the James A. FitzPatrick Nuclear Power Plant", December 1991 (proprietary)."

b. PURPOSE

The change adds a reference.

c. SAFETY IMPLICATIONS

This reference is the PUSAR. It is added to Bases 2.1 to clarify the location of supporting information. It can have no safety significance because it makes no changes.

d. ASSOCIATED CHANGES

Changes A.3, C.1, C.2, C.3, C.6, C.7, C.9, D.2, D.6, F.1 and F.7 are related to this change.

e. REFERENCES

Reference 3

3. Page 35, Bases 3.1.B - Reactor Protection System

a. DESCRIPTION

In the last paragraph, replace the reference "NEDC-31317P" with the reference "NEDC-31317P including latest revision, errata and addenda."

b. PURPOSE

This change revises the reference in the Bases to properly reflect the ECCS-LOCA Analysis for power uprate. Section 4.3 of the PUSAR identifies the analysis performed and its applicability.

c. SAFETY IMPLICATIONS

The purpose of this change is to correct the report in the Bases to reflect the ECCS performance under all loss of coolant accident conditions to satisfy the requirements of 10 CFR 50.46 and 10 CFR 50 Appendix K. The added reference is discussed in Section 4.3 of the PUSAR where it is concluded that the analysis is done using NRC approved methods. This correction has no safety implication.

d. ASSOCIATED CHANGES

Change F.6 is related to this change.

e. REFERENCES

Reference 3, Section 4.3
Reference 5

4. Page 41a, Table 3.1-1 - Reactor Protection System (SCRAM) Instrumentation Requirement

a. DESCRIPTION

Move the trip function "Turbine Stop Valve Closure" from page 42 to the bottom of page 41a.

b. PURPOSE

The trip function was moved from page 42 to the bottom of page 41a to place it with the balance of the trip functions. This is an editorial change made during the processing of the pages.

c. SAFETY IMPLICATIONS

There are no safety implications associated with the movement of text.

d. ASSOCIATED CHANGES

No other change relates to this change.

e. REFERENCES

None

5. Page 188, Bases 3.7 - Primary Containment

a. DESCRIPTION

In the second paragraph, delete the phrase "the limit for complete condensation of."

In the fifth paragraph, delete the word "form" and the value "130°F" and replace them with the word "from" and the value "105°F."

b. PURPOSE

The changes correct typographical errors in the text that were introduced during the amendment process.

The phrase "the limit for complete condensation of," is a repeat of an existing phrase that was added in Amendment 168 . This change corrects this error.

The misspelling of the word "from" was introduced in Technical Specification Amendment 168 and is corrected here.

The value "130°F" was part of the original Technical Specification Bases for restricting the temperature rise in the torus pool during the use of the RCIC, HPCI or relief valves. It reflects the condensation limit for blowdown of 170°F, based on the Humboldt Bay and Bodega Bay tests, less the 40°F pool temperature rise associated with blowdown. Amendment 16 changed this value to "105°F" as part of the amendment that added torus pool temperature limits. These limits addressed containment issues. The 105°F includes the 10°F pool temperature rise over the normal 95°F limit that was allowed for RCIC, HPCI and relief valve testing. The 105°F was inadvertently changed back to 130°F in Amendment 36. This change corrects this error.

c. SAFETY IMPLICATIONS

There are no safety implications associated with the correction of a typographical error.

d. ASSOCIATED CHANGES

No other change relates to this change.

e. REFERENCES

None

6. Page 254-c, Administrative Controls Section 6.9.(A)4.b.2

a. DESCRIPTION

Add the word "revision," after the phrase "NEDC-31317P, October, 1986 including latest."

b. PURPOSE

This change revises the reference used for administrative control to reflect the need to use the latest ECCS-LOCA Analysis. Section 4.3 of the PUSAR identifies the analysis performed for power uprate and its applicability. That NEDC reference is Reference 5 to this safety evaluation.

c. SAFETY IMPLICATIONS

The safety implications are discussed in change F.3.

d. ASSOCIATED CHANGES

Change F.3 relates to this change.

e. REFERENCES

Reference 3, Section 4.3
Reference 5

7. Page 285, Section 7.0 - References

a. DESCRIPTION

Add references "(18) General Electric Report NEDC-32016P, "Power Uprate Safety Analysis for the James A. FitzPatrick Nuclear Power Plant," December 1991 (proprietary)."; "(19) James A. FitzPatrick Calculation JAF-CALC-RAD-00008, "Radiological Consequences of Design Basis Accidents at James A. FitzPatrick," November 1991."; and "(20) General Electric Report GE-NE-187-45-1191P, "Containment Systems Evaluation," (proprietary)."

Delete from Reference (10) the phrase "Progress Report for Period Ending December 31, 1966."

b. PURPOSE

The identified references were used in making changes to the prior pages. The deletion from Reference (10) is administrative. The deleted phrase was repeated twice and represents a typographical error made when adding the reference.

c. SAFETY IMPLICATIONS

There are no safety implications associated with the addition of references or correction of typographical errors.

d. ASSOCIATED CHANGES

Changes A.3, C.1, C.2, C.3, C.6, C.7, C.9, D.2, D.7, F.1 and F.2 are related to this change.

e. REFERENCES

Reference 3
Reference 6
Reference 12

III. EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATION

Operation of the FitzPatrick plant at a thermal power of 2536 MWt will not involve a significant hazards consideration as defined in 10 CFR 50.92, since it would not:

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

   The James A. FitzPatrick nuclear power plant was reviewed for operation at a rated power of 2550 MWt at the time of its operating license, Reference 13. This review was based on the original design of the plant. Since that time, a number of safety issues of a generic and plant specific nature as well as plant modifications have changed the originally reviewed design.

   Generic criteria, methodologies and evaluation scope required to uprate BWRs up to 5% were prepared by General Electric and submitted to the NRC in LTR-1. This was supplemented by the submittal of generic evaluations in LTR-2 to determine: which NRC and industry generic communications were applicable to power uprate and how they should be treated; analytical evaluations that could be generically approved; bounding evaluations of components and equipment, and; the effect of power uprate on safety margin. These generic evaluations are supplemented by plant specific evaluations. The Power Uprate Safety Analysis Report (PUSAR) describes the dependence placed on References 1 and 2, the additional analyses that were performed, the results of these additional analyses and overall conclusions on the safety impacts of power uprate.

   The plant systems and components will be within design limits at power uprate conditions with minor modifications. At uprated power, the power plant will not be operated in a manner that is different from current operations except for limited changes to operating parameters such as primary system pressure, steam flow and feedwater temperature. Setpoints are revised as necessary to reflect new operational conditions and analyses. The ECCS-LOCA analysis using current practices demonstrates compliance with design and regulatory acceptance criteria at uprated power.

   The radiological consequences of accidents have been evaluated using more current methodologies with consistent assumptions and continue to meet acceptance criteria. Compliance with NRC dose criteria using current methodologies is discussed in Section 9.2 of the PUSAR. The effect of power uprate on dose analyses now discussed in the FSAR were qualitatively assessed recognizing that power uprate increases doses in direct proportion to the 4.1% increase in thermal power. An increase of 4.1% to the calculated doses currently identified in FSAR Chapter 14 indicates that a reevaluation using the original methodology would have demonstrated compliance with current NRC dose criteria. A review of Table 14.4-2 indicates that, with the 4.1% increase, offsite doses would be substantially less than NRC allowable values. A review of Table 14.8-1 indicates that, with the 4.1% increase, control room doses would be substantially less than NRC allowables except for the main steam line break (MSLB). However, the MSLB dose would drop well below allowables once the proposed change on allowable coolant activity (reduces the limit by more than a factor of ten) is accounted for.

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

   Operation at uprated power involves no changes to the manner in which the plant is operated. There are changes to operational parameters and setpoints but analyses of these identified no new failure modes or accident scenarios. The effects of transients and accidents fall within design capabilities. Systems and components are capable of operating and performing their safety functions at uprated power. No mechanisms for creating a new or different accident were identified.

3. involve a significant reduction in a margin of safety.

   The power uprate will not result in significant increases to primary system temperature and pressure due to postulated operating transients or accidents. These and other margins of safety have been discussed in the PUSAR, where it is demonstrated that there will be no reductions in the margin of safety because the plant will still meet its design and regulatory acceptance criteria. For example, the core will continue to be operated with the same margin to the safety limit minimum critical power ratio. Fuel thermal limits will continue to meet NRC acceptance criteria. Plant systems and equipment are designed for uprated power conditions and have been evaluated for their capability to perform at uprated conditions. They will continue to perform within design limits.

IV. <u>IMPLEMENTATION OF THE PROPOSED CHANGES</u>

Implementation of the proposed changes will not adversely affect the Fire Protection Program at the FitzPatrick plant. Since there are no plant configuration or combustible load changes, there is no affect on the fire suppression or detecion system. The increase in thermal power will increase the normal source terms in the primary system but these will continue to be well below design values. The impacts on the ALARA program are therefore expected to be minimal. The changes will effect the environment but the impacts will be minimal. There will be no need to change the currently approved radioactive material discharge limits for gaseous and liquid discharges. The increases in radiological levels from the primary system will be proportional to the increase in thermal power. These are restricted to a new and lower limit of 0.2 μCi/gm of dose equivalent I-131. This reduction is more than an order of magnitude. The thermal discharges to the lake will increase slightly and a request to modify the State Pollution Discharge Elimination System (SPDES) is currently planned. The limits in this permit will continue to be met.

V. <u>CONCLUSION</u>

Because the changes will slightly increase the consequences of a power dependent accident, they constitute an unreviewed safety question as defined in 10 CFR 50.59. The dose increases 4.1% with an increase of 4.1% in the thermal power level when analyses are performed using the same methodology.

Operation of the FitzPatrick plant in accordance with the proposed amendment has been assessed in the power uprate safety evaluation, Reference 3, and it has been demonstrated in accordance with 10 CFR 50.92 that the changes would not:

1. involve a significant increase in the probability or consequences of an accident previously evaluated;
2. create the possibility of a new or different kind of accident from any accident previously evaluated; or
3. involve a significant reduction in a margin of safety.

VI. REFERENCES

1. General Electric Licensing Topical Report NEDC-31897P-A "Generic Guidelines for General Electric Boiling Water Reactor Power Uprate," (LTR-1) (proprietary)
2. General Electric Licensing Topical Report NEDC-31984P "Generic Evaluations of General Electric Boiling Water Reactor Power Uprate," July 1991 and Supplement 1, October 1991 (LTR-2) (proprietary)
3. General Electric Report NEDC-32016P "Power Uprate Safety Analysis for the James A. FitzPatrick Nuclear Power Plant," December 1991 (PUSAR) (proprietary)
4. Stone & Webster Engineering Corporation "Core Power Uprate Engineering Report for James A. FitzPatrick Nuclear Power Plant," December 1991 (Engineering Report)
5. General Electric Report NEDC-31317P-1, Revision 1, "James A. FitzPatrick Nuclear Power Plant SAFER/GESTR Loss-of-Coolant Accident Safety Analysis Report," August 1991 (ECCS-LOCA Analysis) (proprietary)
6. James A. FitzPatrick Calculation JAF-CALC-RAD-00008, "Radiological Consequences of Design Basis Accidents at James A. FitzPatrick," November 1991 (Dose Analysis)
7. James A. FitzPatrick calculation JAF 91-002, Revision 1, "Turbine First Stage Pressure Scram Bypass Setpoint (Uprated Condition)," November 1991
8. NYPA letter, J. C. Brons to NRC dated December 20, 1989 (JPN-89-084) regarding proposed changes to the Technical Specifications for S/RV single setpoint performance (JPTS-89-017)
9. General Electric Report NEDC-31697P-1, Revision 1, "Updated S/RV Performance Requirements for the James A. FitzPatrick Nuclear Power Plant," October 1991 (proprietary)
10. NUREG 0123, "Standard Technical Specifications For General Electric Boiling Water Reactors," BWR/4
11. JAF Document No. 22A5747, Revision 1, "Containment Data," 1979
12. General Electric Report GE-NE-187-45-1191, "Containment Systems Evaluation," (proprietary)
13. James A. FitzPatrick Nuclear Power Plant Updated Final Safety Analysis Report
14. James A. FitzPatrick Nuclear Power Plant Safety Evaluation Report (SER), dated November 20, 1972, and Supplements
15. General Electric Report NEDC-24361P, "James A FitzPatrick Nuclear Power Plant Suppression Pool Temperature Response," August 1981 (proprietary)

16. General Electric Report NEDO-30832, "Elimination of Limit on BWR Suppression Pool Temperature For SRV Discharge With Quenchers," December 1984

ATTACHMENT IV to JPN-92-028

GE AFFIDAVIT ON NEDC-32016P

(JPTS-91-025)

New York Power Authority

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333
DPR-59

ATTACHMENT V to JPN-92-028

NEDC-31317P-1
"JAMES A. FITZPATRICK NUCLEAR POWER PLANT SAFER/ GESTR LOSS-OF-COOLANT ACCIDENT SAFETY ANALYSIS REPORT"

(JPTS-91-025)

New York Power Authority

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333
DPR-59

GENERAL ELECTRIC COMPANY

AFFIDAVIT

I, **DAVID J. ROBARE**, being duly sworn, depose and state as follows:

1. I am Manager, Plant Licensing Services, General Electric Company, and have been delegated the function of reviewing the information described in paragraph 2 which is sought to be withheld and have been authorized to apply for its withholding.

2. The information sought to be withheld is contained in General Electric Report NEDC-32016P "Power Uprate Safety Analysis for the James A. FitzPatrick Nuclear Power Plant", dated December 1991. The GE Proprietary portions of this report are identifiable by the "GE Proprietary Information" designation at the top of the page.

3. In designating material as proprietary, General Electric utilizes the definition of proprietary information and trade secrets set forth in the American Law Institute's Restatement of Torts, Section 757. This definition provides:

   "A trade secret may consist of any formula, pattern, device or compilation of information which is used in one's business and which gives him an opportunity to obtain an advantage over competitors who do not know or use it...A substantial element of secrecy must exist, so that, except by the use of improper means, there would be difficulty in acquiring information...Some factors to be considered in determining whether given information is one's trade secret are (1) the extent to which the information is known outside of his business; (2) the extent to which it is known by employees and others involved in his business; (3) the extent of measures taken by him to guard the secrecy of the information; (4) the value of the information to him and to his competitors; (5) the amount of effort or money expanded by him developing the information; (6) the ease or difficulty with which the information could be properly acquired or duplicated by others."

4. Some examples of categories of information which fit into the definition of Proprietary Information are:

   a. Information that discloses a process, method or apparatus where prevention of its use by General Electric's competitors without license from General Electric constitutes a competitive economic advantage over other companies;

   b. Information consisting of supporting data and analyses, including test data, relative to a process, method or apparatus, the application of which provide a competitive economic advantage, e.g., by optimization or improved marketability;

c. Information which if used by a competitor, would reduce his expenditures of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality or licensing of a similar product;

d. Information which reveals cost or price information, production capacities, budget levels or commercial strategies of General Electric, its customers or suppliers;

e. Information which reveals aspects of past, present or future General Electric customer-funded development plans and programs of potential commercial value to General Electric;

f. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection;

g. Information which General Electric must treat as proprietary according to agreements with other parties.

5. Initial approval of proprietary treatment of a document is typically made by the Subsection Manager of the originating component, the person who is most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within the Company is limited on a "need to know" basis and such documents are clearly identified as proprietary.

6. The procedure for approval of external release of such a document typically requires review by the Subsection Manager, Project Manager, Principal Scientist or other equivalent authority, by the Subsection Manager of the cognizant Marketing function (or delegate) and by the Legal Operation for technical content, competitively effect and determination of the accuracy of the proprietary designation in accordance with the standards enumerated above. Disclosures outside General Electric are generally limited to regulatory bodies, customers and potential customers and their agents, suppliers and licensees then only with appropriate protection by applicable regulatory provisions or proprietary agreements.

7. The document mentioned in paragraph 2 above has been evaluated in accordance with the above criteria and procedures and has been found to contain information which is proprietary and which is customarily held in confidence by General Electric.

8. The information to the best of my knowledge and belief has consistently been held in confidence by General Electric Company, no public disclosure has been made, and it is not available in public sources.

GENERAL ELECTRIC COMPANY

AFFIDAVIT

8. All disclosures to third parties have been made pursuant to regulatory provisions of proprietary agreements which provide for maintenance of the information in confidence.

9. Public disclosure of the information sought to be withheld is likely to cause substantial harm to the competitive position of the General Electric Company and deprive or reduce the availability of profit making opportunities. A substantial effort has been expended by General Electric to develop this information.

GENERAL ELECTRIC COMPANY

AFFIDAVIT

STATE OF CALIFORNIA)
) ss:
COUNTY OF SANTA CLARA)

David J. Robare, being duly sworn, deposes and says:

That he has read the foregoing affidavit and the matters stated therein are truly and correct to the best of his knowledge, information, and belief.

Executed at San Jose, California, this 23RD day of DECEMBER 19 91.

David J. Robare

David J. Robare
General Electric Company

Subscribed and sworn before me this 23rd day of December 19 91.

OFFICIAL SEAL
MARY L KENDALL
Notary Public-California
SANTA CLARA COUNTY
My Comm. Exp. Mar. 26, 1993

Mary L. Kendall

Notary Public, State of California

ATTACHMENT VI to JPN-92-028

GE AFFIDAVIT ON NEDC-31317P-1

(JPTS-91-025)

New York Power Authority

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333
DPR-59

ATTACHMENT VII to JPN-92-028

MARK-UP OF CURRENT TECHNICAL SPECIFICATIONS

(JPTS-91-025)

New York Power Authority

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333
DPR-59

1.0 (cont'd)

C. <u>Cold Condition</u> - Reactor coolant temperature $\leq$212°F.

D. <u>Hot Standby Condition</u> - Hot Standby condition means operation with coolant temperature > 212°F, the Mode Switch in Startup/Hot Standby and reactor pressure < ~~1,000~~ 1,040 psig.

E. <u>Immediate</u> - Immediate means that the required action will be initiated as soon as practicable considering the safe operation of the unit and the importance of the required action.

F. <u>Instrumentation</u>

1. Functional Test - A functional test is the manual operation or initiation of a system, subsystem, or component to verify that it functions within design tolerances (e.g., the manual start of a core spray pump to verify that it runs and that it pumps the required volume of water).

2. Instrument Channel Calibration - An instrument channel calibration means the adjustment of an instrument signal output so that it corresponds, within acceptable range, and accuracy, to a known value(s) of the parameter which the instrument monitors. Calibration shall encompass the entire instrument channel including actuation, alarm or trip.

3. Instrument Channel - An instrument channel means an arrangement of a sensor and auxiliary equipment required to generate and transmit to a trip system a single trip signal related to the plant parameter monitored by that instrument channel.

4. Instrument Check - An instrument check is a qualitative determination of acceptable operability by observation of instrument behavior during operation. This determination shall include, where possible, comparison of the instrument with other independent instruments measuring the same variable.

5. Instrument Channel Functional Test - An instrument channel functional test means the injection of a simulated signal into the instrument primary sensor where possible to verify the proper instrument channel response, alarm and/or initiating action.

6. Logic System Function Test - A logic system functional test means a test of relays and contacts of a logic circuit from sensor to activated device to ensure components are operable per design intent. Where practicable, action will go to completion: i.e., pumps

1.0 (cont'd)

opened to perform necessary operational activities.

2. At least one door in each airlock is closed and sealed.

3. All automatic containment isolation valves are operable or de-activated in the isolated position.

4. All blind flanges and manways are closed.

N. Rated Power - Rated power refers to operation at a reactor power of ~~2,436~~ 2,536 MWt. This is also termed 100 percent power and is the maximum power level authorized by the operating license. Rated steam flow, rated coolant flow, rated nuclear system pressure, refer to the values of these parameters when the reactor is at rated power (Reference 1).

O. Reactor Power Operation - Reactor power operation is any operation with the Mode Switch in the Startup/Hot Standby or Run position with the reactor critical and above 1 percent rated thermal power.

P. Reactor Vessel Pressure - Unless otherwise indicated, reactor vessel pressures listed in the Technical Specifications are those measured by the reactor vessel steam space sensor.

Q. Refueling Outage - Refueling outage is the period of time between the shutdown of the unit prior to refueling and the startup of the Plant subsequent to that refueling.

R. Safety Limits - The safety limits are limits within which the reasonable maintenance of the fuel cladding integrity and the reactor coolant system integrity are assured. Violation of such a limit is cause for unit shutdown and review by the Atomic Energy Commission before resumption of unit operation. Operation beyond such a limit may not in itself result in serious consequences but it indicates an operational deficiency subject to regulatory review.

S. Secondary Containment Integrity - Secondary containment integrity means that the reactor building is intact and the following conditions are met:

1. At least one door in each access opening is closed.

2. The Standby Gas Treatment System is operable.

3. All automatic ventilation system isolation valves are operable or secured in the isolated position.

T. Surveillance Frequency - Periodic

Z. Top of Active Fuel

The Top of Active Fuel, corresponding to the top of the enriched fuel column of each fuel bundle, is located 352.5 inches above vessel zero, which is the lowest point in the inside bottom of the reactor vessel. (See General Electric drawing No. 919D690BD.)

AA. Rod Density

Rod density is the number of control rod notches inserted expressed as a fraction of the total number of control rod notches. All rods fully inserted is a condition representing 100 percent rod density.

AB. Purge-Purging

Purge or Purging is the controlled process of discharging air or gas from a confinement in such a manner that replacement air or gas is required to purify the confinement.

AC. Venting

Venting is the controlled process of releasing air or gas from a confinement in such a manner that replacement air or gas is not provided or required.

AD. Core Operating Limits Report (COLR)

This report is the plant-specific document that provides the core operating limits for the current operating cycle. These cycle-specific operating limits shall be determined for each reload cycle in accordance with Specification 6.9.A.4. Plant operation within these operating limits is addressed in individual Technical Specifications.

Insert 1

INSERT 1

AE. References

1. General Electric Report NEDC-32016P, "Power Uprate Safety Analysis for the James A. FitzPatrick Nuclear Power Plant," December 1991 (proprietary).

2.1 (cont'd)

2. Reactor Water Low Level Scram Trip Setting

Reactor low water level scram setting shall be $\geq$177 in. above the top of the active fuel (TAF) at normal operating conditions.

3. Turbine Stop Valve Closure Scram Trip Setting

Turbine stop valve scram shall be $\leq$ 10 percent valve closure from full open when ~~above 217 psig turbine first stage pressure.~~ the reactor is at or above 29% of rated power.

4. Turbine Control Valve Fast Closure Scram Trip Setting

Turbine control valve fast closure scram control oil pressure shall be set at 500 < P < 850 psig.

5. Main Steam Line Isolation Valve Closure Scram Trip Setting

Main steam line isolation valve closure scram shall be $\leq$ 10 percent valve closure from full open.

6. Main Steam Line Isolation Valve Closure on Low Pressure

When in the run mode main steam line low pressure initiation of main steam line isolation valve closure shall be $\geq$825 psig.

BASES

2.1 FUEL CLADDING INTEGRITY

The abnormal operational transients applicable to operation of the FitzPatrick Unit have been analyzed throughout the spectrum of planned operating conditions up to the thermal power condition of ~~2436~~ 2,536 MWt. The analyses were based upon plant operation in accordance with the operating map given in the current load line limit analysis. In addition, ~~2436~~ 2,536 MWt is the licensed maximum power level of FitzPatrick, and this represents the maximum steady-state power which shall not knowingly be exceeded.

The transient analyses performed for each reload are ~~given~~ described in Reference 2. Models and model conservatism are ~~also~~ described in this reference. As discussed in Reference 4, the core wide transient analysis for one recirculation pump operation is conservatively bounded by two-loop operation analysis, and the flow-dependent rod block and scram setpoint equations are adjusted for one-pump operation. Insert 2

Fuel cladding integrity is assured by the applicable operating limit MCPR for steady state conditions given in the Core Operating Limits Report (COLR). These operating limit MCPR's are derived from the established fuel cladding integrity Safety Limit, and an analysis of abnormal operational transients. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient.

The most limiting transients have been analyzed to determine which result in the largest reduction in CRITICAL POWER RATIO. The type of transients evaluated were increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest delta MCPR. When added to the Safety Limit, the required operating limit MCPR in the Core Operating Limits Report is obtained.

The evaluation of a given transient begins with the system initial parameters shown in the current reload analysis and Reference 2 that are input to the core dynamic behavior transient computer programs described in Reference 2. The output of these programs along with the initial MCPR form the input for the further analyses of the thermally limited bundle with a single channel transient thermal hydraulic code. The principal result of the evaluation is the reduction in MCPR caused by the transient.

INSERT 2

Reference 1 evaluates the safety significance of uprated power operation at 2,536 MWt. This evaluation is consistent with and demonstrates the acceptability of the transient analyses required by Reference 2.

2.1 BASES (Cont'd)

The MCPR operating limits in the COLR are conservatively assumed to exist prior to initiation of the transients.

This choice of using conservative values of controlling parameters and initiating transients at the design power level, produces more pessimistic answers than would result by using expected values of control parameters and analyzing at higher power levels.

Steady-state operation without forced recirculation is not permitted. The analysis to support operation at various power and flow relationships has considered operation with either one or two recirculation pumps.

In summary:

- The abnormal operational transients were analyzed to the licensed maximum power level.
- The licensed maximum power level is ~~2436~~ 2,536 MWt.
- Analyses of transients employ adequately conservative values of the controlling reactor parameters.
- The analytical procedures now used result in a more logical answer than the alternative method of assuming a higher starting power in conjunction with the expected values for the parameters.

A. Trip Settings

The bases for individual trip settings are discussed in the following paragraphs.

1. Neutron Flux Trip Settings

a. IRM Flux Scram Trip Setting

The IRM system consists of 8 chambers, 4 in each of the reactor protection system logic channels. The IRM is a 5-decade instrument which covers the range of power level between that covered by the SRM and the APRM. The 5 decades are covered by the IRM by means of a range switch and the 5 decades are broken down into 10 ranges, each being one-half of a decade in size. The IRM scram trip setting of 120 divisions is active in each range of the IRM. For example, if the instrument were on Range 1, the scram setting would be a 120 divisions for that range; likewise, if the instrument were on range 5, the scram would be 120 divisions on that range. Thus, as the IRM is ranged up to accommodate the increase in power level, the scram trip setting is also ranged up. The most significant sources of reactivity change during the power increase are due to control rod withdrawal. For insequence control rod withdrawal, the rate of change of power is slow enough due to the physical limitation of withdrawing control rods, that heat flux is in equilibrium with the neutron flux and an IRM scram would result in a reactor shutdown well before any Safety Limit is exceeded.

3. Turbine Stop Valve Closure Scram Trip Settings

The turbine stop valve closure scram trip anticipates the pressure, neutron flux and heat flux increase that could result from rapid closure of the turbine stop valves. With a scram trip setting of ≤ 10 percent of valve closure from full open, the resultant increase in surface heat flux is limited such that MCPR remains along the Safety Limit even during the worst case transient that assumes the turbine bypass is closed. This scram is bypassed when ~~turbine steam flow is below 30%~~ reactor power is below 29% of rated, as measured by turbine first stage pressure, consistent with the safety analysis discussed in Reference 1.

4. Turbine Control Valve Fast Closure Scram Trip Setting

This turbine control valve fast closure scram anticipates the pressure, neutron flux, and heat flux increase that could result from fast closure of the turbine control valves due to load rejection exceeding the capability of the turbine bypass. The Reactor Protection System initiates a scram when fast closure of the control valves is initiated by the fast acting solenoid valves. This is achieved by the action of the fast acting solenoid valves in rapidly reducing hydraulic control oil pressure at the main turbine control valve actuator disc dump valves. This loss of pressure is sensed by pressure switches whose contacts form the one-out-of-two-twice logic input to the reactor protection system. This trip setting, a nominally 50 percent greater closure time and a different valve characteristic from that of the turbine stop valve, combine to produce transients very similar and no more severe than for the stop valve. No significant change in MCPR occurs. Relevant transient analyses are discussed in Section 14.5 of the Final Safety Analysis Report and Reference 1. This scram is bypassed when ~~turbine steam flow is below 30 percent~~ reactor power is below 29 percent of rated, as measured by turbine first stage pressure.

5. Main Steam Line Isolation Valve Closure Scram Trip Setting

The low pressure isolation of the main steam lines at 825 psig was provided to give protection against rapid reactor depressurization and the resulting rapid cooldown of the vessel. Advantage was taken of the scram feature which occurs when the main steam line isolation valves are closed, to provide for reactor shutdown so that high power operation at low reactor pressure does not occur, thus providing protection for the fuel cladding integrity safety limit. Operation of the reactor at pressures lower than 825 psig requires that the Reactor Mode Switch be in the Startup position where protection of the fuel cladding integrity safety limit is provided by the APRM high neutron flux scram and the IRM. Thus, the combination of main steam line low pressure isolation and isolation valve closure scram assures the availability of neutron flux scram protection over the entire range of applicability of the fuel cladding integrity safety limit. In addition, the isolation valve closure scram anticipates the pressure and flux transients which occur during normal or inadvertent isolation valve closure. With the scrams set at ≤ 10 percent valve closure, there is no increase in neutron flux.

6. Main Steam Line Isolation Valve Closure on Low Pressure

The low pressure isolation minimum limit at 825 psig was provided to give protection against fast reactor depressurization and the resulting rapid cooldown of the vessel. Advantage was taken of the scram feature which occurs when the main steam line isolation valves are closed to provide for reactor shutdown so that operation at pressures lower than those specified in the thermal hydraulic safety limit does not occur, although operation at a pressure lower than 825 psig would not necessarily constitute an unsafe condition.

2.1 BASES (Cont'd)

C. References

1. (Deleted) R.
2. "General Electric Standard Application for Reactor Fuel", NEDE 24011-P-A (Approved revision number applicable at time that reload fuel analyses are performed).
3. (Deleted)
4. FitzPatrick Nuclear Power Plant Single-Loop Operation, NEDO-24281, August, 1980.

Insert 3

INSERT 3

General Electric Report, NEDC-32016P, "Power Uprate Safety Analysis for the James A. FitzPatrick Nuclear Power Plant", December 1991 (proprietary).

1.2 REACTOR COOLANT SYSTEM

APPLICABILITY:

Applies to limits on reactor coolant system pressure.

OBJECTIVE:

To establish a limit below which the integrity of the Reactor Coolant System is not threatened due to an overpressure condition.

SPECIFICATION:

1. The reactor coolant system pressure shall not exceed 1,325 psig at any time when irradiated fuel is present in the reactor vessel.

2.2 REACTOR COOLANT SYSTEM

APPLICABILITY:

Applies to trip settings of the instruments and devices which are provided to prevent the reactor coolant system safety limits from being exceeded.

OBJECTIVE:

To define the level of the process variables at which automatic protective action is initiated to prevent the safety limits from being exceeded.

SPECIFICATION:

1. The Limiting Safety System setting shall be specified below:

   A. Reactor coolant high pressure scram shall be ≤ ~~1,045~~ 1,080 psig.

   B. Reactor coolant system safety/relief valve nominal settings shall be ~~as~~ ≤ 1,145 psig ~~follows:~~

   Safety/Relief Valves

   2 valves at 1090 psig
   2 valves at 1105 psig
   7 valves at 1140 psig

   The allowable setpoint error for each safety/relief valve shall be ± 1 percent.

1.2 and 2.2 BASES

The reactor coolant pressure boundary integrity is an important barrier in the prevention of uncontrolled release of fission products. It is essential that the integrity of this boundary be protected by establishing a pressure limit to be observed for all operating conditions and whenever there is irradiated fuel in the reactor vessel.

The pressure safety limit of 1,325 psig as measured by the vessel steam space pressure indicator is equivalent to 1,375 psig at the lowest elevation of the Reactor Coolant System. The 1,375 psig value is derived from the design pressures of the reactor pressure vessel and reactor coolant system piping. The respective design pressures are 1250 psig at 575°F for the reactor vessel, 1148 psig at 568°F for the recirculation suction piping and 1274 psig at 575° for the discharge piping. The pressure safety limit was chosen as the lower of the pressure transients permitted by the applicable design codes: 1965 ASME Boiler and Pressure Vessel Code, Section III for pressure vessel and 1969 ANSI B31.1 Code for the reactor coolant system piping. The ASME Boiler and Pressure Vessel Code permits pressure transients up to 10 percent over design pressure (110% x 1,250 - 1,375 psig) and the ANSI Code permits pressure transients up to 20 percent over the design pressure (120% x 1,150 - 1,380 psig). The safety limit pressure of 1,375 psig is referenced to the lowest elevation of the Reactor Coolant System.

The current reload analysis shows that the main steam isolation valve closure transient, with flux scram, is the most severe event resulting directly in a reactor coolant system pressure increase. The reactor vessel pressure code limit of 1,375 psig, given in FSAR Section 4.2, is above the peak pressure produced by the event above. Thus, the pressure safety limit (1,375 psig) is well above the peak pressure that can result from reasonably expected overpressure transients. (See current reload analysis for the curve produced by this analysis.) Reactor pressure is continuously indicated in the control room during operation.

A safety limit is applied to the Residual Heat Removal System (RHRS) when it is operating in the shutdown cooling mode. When operating in the shutdown cooling mode, the RHRS is included in the reactor coolant system.

~~The numerical distribution of safety/relief valve setpoints shown in 2.2.1.B (3 @ 1090 psi, 2 @ 1105 psi, 7 @ 1140 psi) is justified by analyses described in the General Electric report NEDO-24129-1, Supplement 1,~~ and assures that the structural acceptance criteria set forth in the Mark I Containment Short Term Program are satisfied.

Insert 4

Amendment No. ~~58~~, ~~84~~, 184

INSERT 4

The numerical safety/relief valve setpoint shown in 2.2.1.B is justified by analyses described in the General Electric report NEDC-32016P.

3.1 BASES (cont'd)

is discharged from the reactor by a scram can be accommodated in the discharge piping. Each scram discharge instrument volume accommodates in excess of 34 gallons of water and is the low point in the piping. No credit was taken for this volume in the design of the discharge piping as concerns the amount of water which must be accommodated during a scram.

During normal operation the discharge volume is empty, however, should it fill with water, the water discharged to the piping from the reactor could not be accommodated, which would result in slow scram times or partial control rod insertion. To preclude this occurrence, level detection instruments have been provided in each instrument volume which alarm and scram the reactor when the volume of water reaches 34.5 gallons. As indicated above, there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or amount of insertion of the control rods. This function shuts the reactor down while sufficient volume remains to accommodate the discharged water and precludes the situation in which a scram would be required but not be able to perform its function adequately.

A Source Range Monitor (SRM) System is also provided to supply additional neutron level information during startup but has no scram functions (reference paragraph 7.5.4 FSAR).

The IRM high flux and APRM ≤ 15% power scrams provide adequate coverage in the startup and intermediate range. Thus, the IRM and APRM systems are required to be operable in the refuel and startup/hot standby modes. The APRM ≤ 120% power and flow referenced scrams provide required protection in the power range (reference FSAR Section 7.5.7). The power range is covered only by the APRMs. Thus, the IRM system is not required in the run mode.

The high reactor pressure, high drywell pressure, reactor low water level and scram discharge volume high level scrams are required for startup and run modes of plant operation. They are, therefore, required to be operational for these modes of reactor operation.

The requirement to have the scram functions indicated in Table 3.1-1 operable in the refuel mode assures that shifting to the refuel mode during reactor power operation does not diminish the protection provided by the Reactor Protection System.

Turbine stop valve closure occurs at 10 percent of valve closure. Below ~~217 psig turbine first stage pressure (30 percent of rated)~~, the scram signal due to turbine stop valve closure is bypassed because the flux and pressure scrams are adequate to protect the reactor.

29% of rated reactor power

3.1 BASES (cont'd)

Turbine control valves fast closure initiates a scram based on pressure switches sensing electro-hydraulic control (EHC) system oil pressure. The switches are located between fast closure solenoids and the disc dump valves, and are set relative (500<P<850 psig) to the normal (EHC) oil pressure of 1,600 psig so that based on the small system volume, they can rapidly detect valve closure or loss of hydraulic pressure.

The requirement that the IRM's be inserted in the core when the APRM's read 2.5 indicated on the scale in the start-up and refuel modes assures that there is proper overlap in the neutron monitoring system functions and thus, that adequate coverage is provided for all ranges of reactor operation.

B. The limiting transient which determines the required steady state MCPR limit depends on cycle exposure. The operating limit MCPR values as determined from the transient analysis in the current reload submittal for various core exposures are specified in the Core Operating Limits Report (COLR).

The ECCS performance analyses assumed reactor operation will be limited to MCPR = 1.20, as described in NEDO-21662 and NEDC-31317P. The Technical Specifications limit operation of the reactor to the more conservative MCPR based on consideration of the limiting transient as specified in the COLR.

including latest revision, errata and addenda

JAFNPP

TABLE 3.1-1 (cont'd)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Minimum No. of Operable Instrument Channels per Trip System (1)	Trip Function	Trip Level Setting 1	Modes in Which Function Must be Operable			Total Number of Instrument Channels Provided by Design for Both Trip Systems	Action (1)
			Refuel (6)	Startup	Run		
2	APRM Downscale	≥ 2.5 indicated on scale (9)			X	6 Instrument Channels	A or B
2	High Reactor Pressure	≤ ~~1045~~ psig ≤1,080	X(8)	X	X	4 Instrument Channels	A
2	High Drywell Pressure	≤ 2.7 psig	X(7)	X(7)	X	4 Instrument Channels	A
2	Reactor Low Water Level	≥177 in. above TAF	X	X	X	4 Instrument Channels	A
3	High Water Level in Scram Discharge Volume	≤ 34.5 gallons per Instrument Volume	X(2)	X	X	8 Instrument Channels	A
2	Main Steam Line High Radiation	≤ 3x normal full power background (16)	X	X	X	4 Instrument Channels	A
4	Main Steam Line Isolation Valve Closure	≤10% valve closure			X(5)	8 Instrument Channels	A
4	Turbine Stop Valve Closure	≤10% valve closure			X(4)(5)	8 Instrument Channels	A or C

Amendment No. ~~15~~, ~~43~~, ~~67~~, ~~78~~, ~~87~~, ~~96~~, ~~129~~, ~~132~~

TABLE 3.1-1 (cont'd)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Minimum No. of Operable Instrument Channels per Trip System (1)	Trip Function	Trip Level Setting	Modes in Which Function Must be Operable: Refuel (6)	Startup	Run	Total Number of Instrument Channels Provided by Design for Both Trip Systems	Action (1)
4	Turbine Stop Valve Closure	≤ 10% valve closure			X(4)(5)	8 Instrument Channels	A or C

Move Table entry to bottom of page 41a

NOTES OF TABLE 3.1-1

1. There shall be two operable or tripped trip systems for each function, except as specified in 4.1.D. From and after the time that the minimum number of operable instrument channel for a trip system cannot be met, that affected trip system shall be placed in the safe (tripped) condition, or the appropriate actions listed below shall be taken.

   A. Initiate insertion of operable rods and complete insertion of all operable rods within four hours.

   B. Reduce power level to IRM range and place Mode Switch in the Startup Position within eight hours.

   C. Reduce power to less than ~~30~~ <29% percent of rated.

2. Permissible to bypass, if Refuel and Shutdown positions of the Reactor Mode Switch.

3. Deleted.

4. Bypassed ~~when turbine first stage pressure is less than 217 psig or~~ the reactor power is less than ~~30~~ 29% percent of rated.

5. The design permits closure of any two lines without a scram being initiated.

6. When the reactor is subcritical and the reactor water temperature is less than 212°F, only the following trip functions need to be operable:
   A. Mode Switch in Shutdown
   B. Manual Scram

Amendment No. ~~55~~, ~~91~~, 111, 134

3.5 (Cont'd)

DELETED

C. HIGH PRESSURE COOLANT INJECTION (HPCI SYSTEM)

1. The HPCI System shall be operable whenever the reactor pressure is greater than 150 psig and reactor coolant temperature is greater than 212°F and irradiated fuel is in the reactor vessel, except as specified below:

4.5 (Cont'd)

C. HIGH PRESSURE COOLANT INJECTION (HPCI SYSTEM)

Surveillance of HPCI System shall be performed as follows provided a reactor steam supply is available. If steam is not available at the time the surveillance test is scheduled to be performed, the test shall be performed within 10 days of continuous operation from the time steam becomes available.

1. HPCI System testing shall be as specified in 4.5.A.1.a, b, c, d, f, and g except that the HPCI pump shall deliver at least 4,250 gpm against a system head corresponding to a reactor vessel pressure of ~~1,120~~ 1,195 psig to 150 psig.

3.5 (cont'd)

4.5 (cont'd)

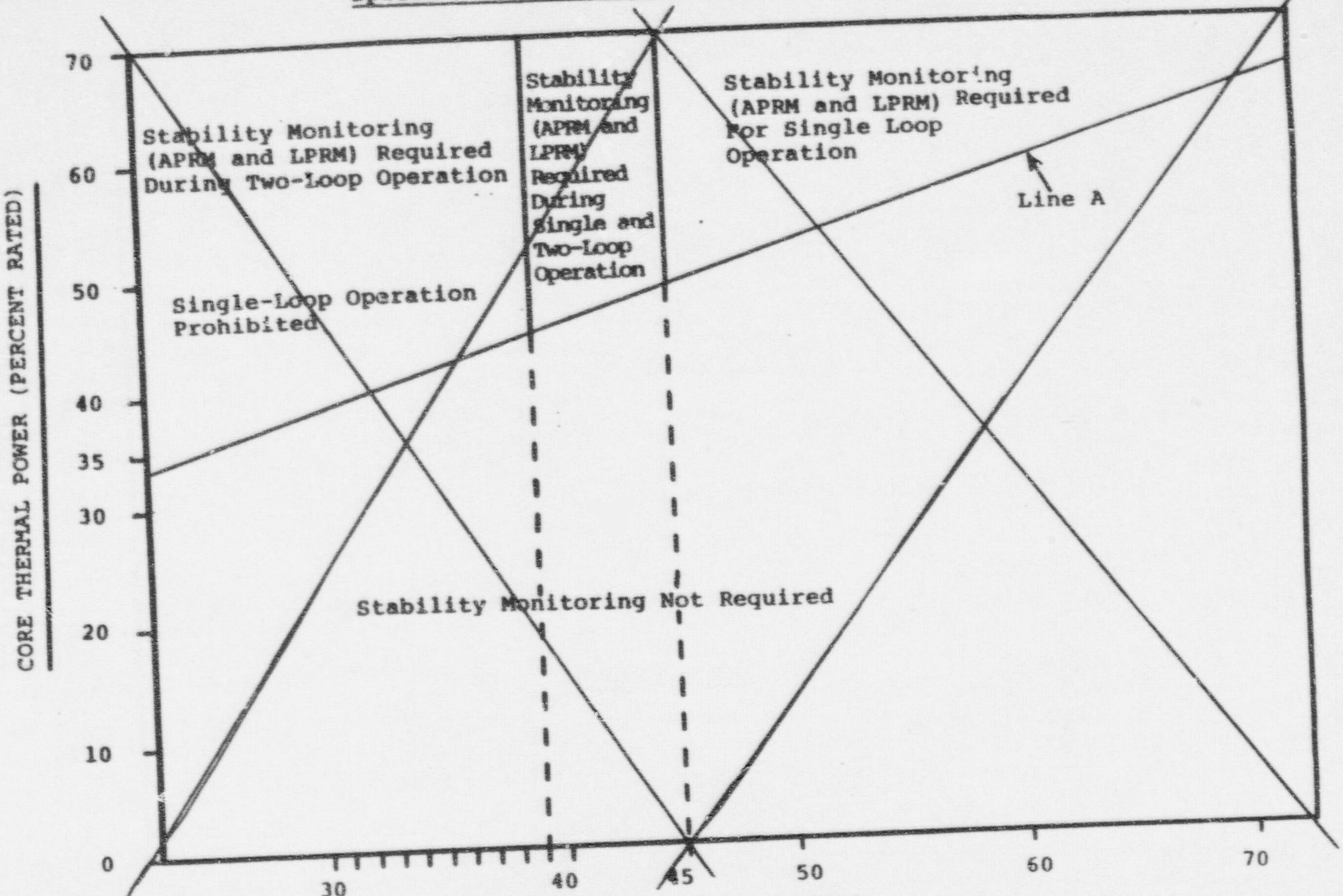
The RCIC pump shall deliver at least 400 gpm for a system head corresponding to a reactor pressure of ~~1,120~~ 1,195 psig to 150 psig.

2. When it is determined that the RCIC System is inoperable at a time when it is required to be operable, the HPCI System shall be verified to be operable immediately and daily thereafter.

Insert 5

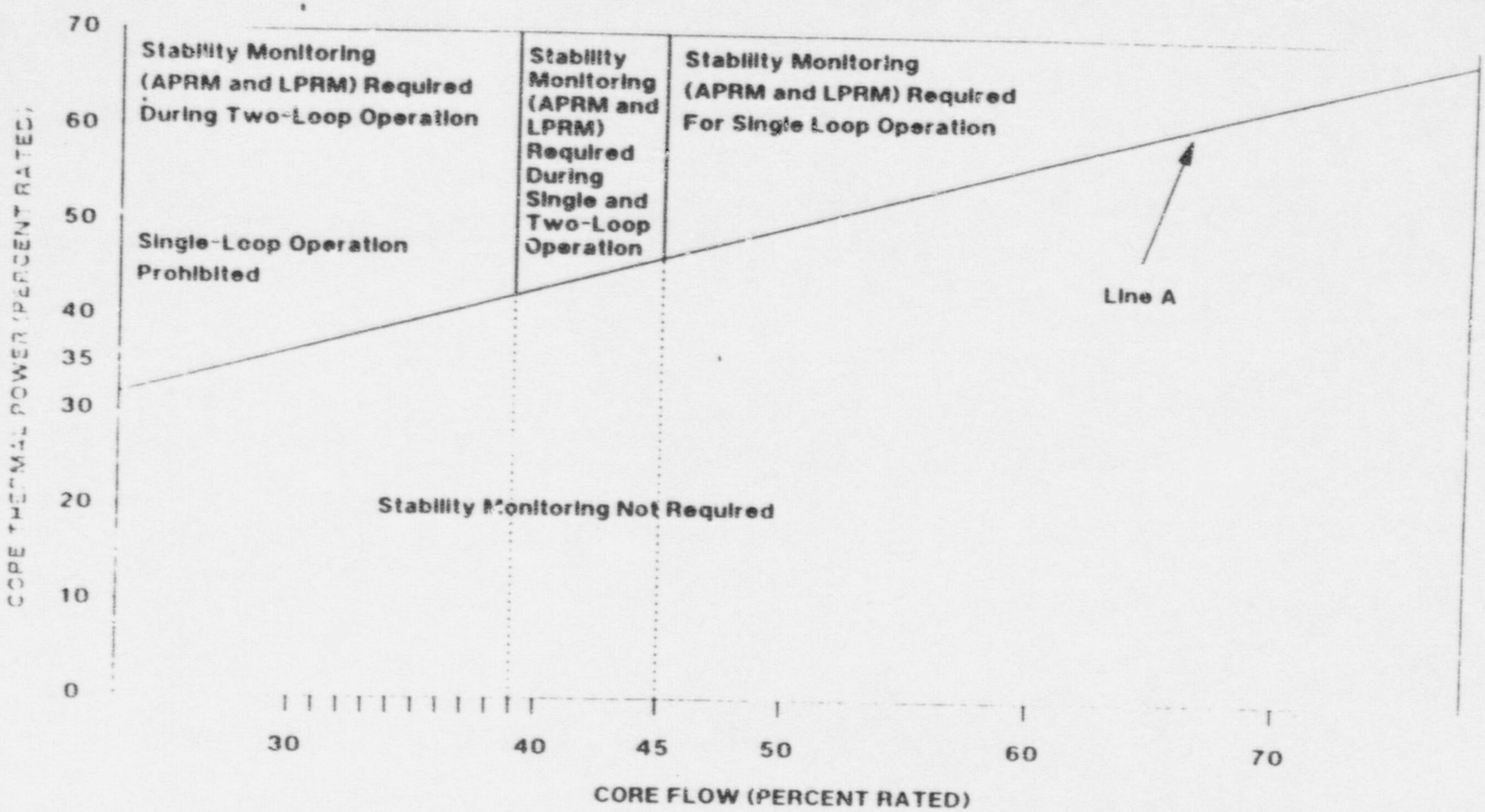
Figure 3.5-1

Thermal Power and Core Flow Limits of Specifications 3.5.J.1, 3.5.J.2 and 3.5.J.3



Insert 5

Figure 3.5-1
Thermal Power and Core Flow Limits of Specifications 3.5.J.1, 3.5.J.2, and 3.5.J.3



Amendment No. ~~14~~. ~~30~~. ~~52~~. ~~64~~. ~~98~~.

3.6 (cont'd)

B. Deleted

C. Coolant Chemistry

1. The reactor coolant system radioactivity concentration in water shall not exceed the equilibrium value of ~~0.1~~ 0.2 μCi/gm of dose equivalent I-131. This limit may be exceeded, following a power transient, for a maximum of 48 hr. During this iodine activity transient the iodine concentrations shall not exceed the equilibrium limits by more than a factor of 10 whenever the main steamline isolation valves are open. The reactor shall not be operated more than 5 percent of its annual power operation under this exception to the equilibrium limits. If the iodine concentration exceeds the equilibrium limit by more than a factor of 10, the reactor shall be placed in a cold condition within 24 hr.

4.6 (cont'd)

7. Reactor Vessel Flux Monitoring

The reactor vessel Flux Monitoring Surveillance Program complies with the intent of the May, 1983 revision to 10 CFR 50, Appendices G and H. The next flux monitoring surveillance capsule shall be removed after 15 effective full power years (EFPYs) and the test procedures and reporting requirements shall meet the requirements of ASTM E 185-82.

B. Deleted

C. Coolant Chemistry

1. a. A sample of reactor coolant shall be taken at least every 96 hr and analyzed for gross gamma activity.

   b. Isotopic analysis of a sample of reactor coolant shall be made at least once/month.

   c. A sample of reactor coolant shall be taken prior to startup and at 4 hr intervals during startup and analyzed for gross gamma activity.

   d. During plant steady state operation and following an offgas activity increase (at the Steam Jet Air Ejectors) of 10,000 μCi/sec within a 48 hr. period or a power level change of $\geq$20 percent of full rated power/hr reactor coolant samples shall be taken and analyzed for gross gamma activity. At least three samples will be taken at 4 hr intervals. These sampling requirements may be omitted whenever the equilibrium I-131 concentration in the reactor coolant is less than 0.007 μCi/ml.

3.6 and 4.6 BASES (cont'd)

The expected neutron fluence at the reactor vessel wall can be determined at any point during plant life based on the linear relationship between the reactor thermal power output and the corresponding number of neutrons produced. Accordingly, neutron flux wires were removed from the reactor vessel with the surveillance specimens to establish the correlation at the capsule location by experimental methods. The flux distribution at the vessel wall and 1/4 thickness (1/4T) depth was analytically determined as a function of core height and azimuth to establish the peak flux location in the vessel and the lead factor of the surveillance specimens.

Regulatory Guide 1.99, Revision 2 is used to predict the shift in RT_{NDT} as a function of fluence in the reactor vessel beltline region. An evaluation of the irradiated surveillance specimens, which were withdrawn from the reactor in April, 1985 (6 EFPY), shows a shift in RT_{NDT} less than that predicted by Regulatory Guide 1.99, Revision 2.

Operating limits for the reactor vessel pressure and temperature during normal heatup and cooldown, and during in-service hydrostatic and leak testing were established using 10 CFR 50 Appendix G, May, 1983 and Appendix G of the Summer 1984 Addenda to Section III of the ASME Boiler and Pressure Vessel Code. These operating limits assure that the vessel could safely accommodate a postulated surface flaw having a depth of 0.24 inch at the flange-to-vessel junction, and one-quarter of the material thickness at all other reactor vessel locations and discontinuity regions. For the purpose of setting these operating limits, the reference temperature, RT_{NDT}, of the vessel material was estimated from impact test data taken in accordance with the requirements of the Code to which the vessel was designed and manufactured (1965 Edition including Winter 1966 addenda). The RT_{NDT} values for the reactor vessel flange region and for the reactor vessel shell beltline region are 30°F, based on fabrication test reports. The RT_{NDT} for the remainder of the vessel is 40°F.

The first surveillance capsule containing test specimens was withdrawn in April, 1985 after 6 EFPY. The test specimens removed were tested according to ASTM E 185-82 and the results are in GE report MDE-49-0386. The next surveillance capsule will be removed after 15 EFPYs of operation and the results of the examination used as a basis for revision of Figure 3.6-1 curves A, B and C for operation of the plant after 16 EFPYs.

Figure 3.6-1 is comprised of three parts: Part 1, Part 2, and Part 3. Parts 1, 2, and 3 establish the pressure-temperature limits for plant operations through 12, 14, and 16 Effective Full Power Years (EFPY) respectively. The appropriate figure and the pressure-temperature curves are dependent on the number of accumulated EFPY. Figure 3.6-1, Part 1 is for operation through 12 EFPY, Figure 3.6-1, Part 2 is for operation at greater than 12 EFPY through 14 EFPY, and Figure 3.6-1, Part 3 is for operation at greater than 14 EFPY through 16 EFPY. The curves contained in Figure 3.6-1 are developed from the General Electric Report DRF 137-0010, "Implementation of Regulatory Guide 1.99, Revision 2 for the James A. FitzPatrick Nuclear Power Plant," dated June, 1989.

Figure 3.6-1 curve A establishes the minimum temperature for hydrostatic and leak testing required by the ASME Boiler and Pressure Vessel Code, Section XI. Test pressures for in-service hydrostatic and leak testing are a function of the testing temperature and the component material. Accordingly, the maximum hydrostatic test pressure will be 1.1 times the operating pressure or about ~~1105~~ 1,144 psig.

3.6 and 4.6 BASES (cont'd)

B. Deleted

C. Coolant Chemistry

A radioactivity concentration limit of 20 μCi/ml total iodine can be reached if the gaseous effluents are near the limit as set forth in Radiological Effluent Technical Specification Section 3.2.a if there is a failure or a prolonged shutdown of the cleanup demineralizer.

Insert 6

~~In the event of a steam line rupture outside the drywell, with this coolant activity level, the resultant radiological dose at the site boundary would be 33 rem to the thyroid, under adverse meteorological conditions assuming no more than 3.1 μCi/gm of dose equivalent I-131.~~ The reactor water sample will be used to assure that the limit of Specification 3.6.C is not exceeded. The total radioactive iodine activity would not be expected to change rapidly over a period of 96 hr. In addition, the trend of the stack offgas release rate, which is continuously monitored, is a good indicator of the trend of the iodine activity in the reactor coolant. Also during reactor startups and large power changes which could affect iodine levels, samples of reactor coolant shall be analyzed to insure iodine concentrations are below allowable levels. Analysis is required whenever the I-131 concentration is within a factor of 100 of its allowable equilibrium value. The necessity for continued sampling following power and offgas transients will be reviewed within 2 years of initial plant startup.

The surveillance requirements 4.6.C.1 may be satisfied by a continuous monitoring system capable of determining the total iodine concentration in the coolant on a real time basis, and annunciating at appropriate concentration levels such that sampling for isotopic analysis can be initiated. The design details of such a system must be submitted for evaluation and accepted by the Commission prior to its implementation and incorporation in these Technical Specifications.

Since the concentration of radioactivity in the reactor coolant is not continuously measured, coolant sampling would be ineffective as a means to rapidly detect gross fuel element failures. However, some capability to detect gross fuel element failures is inherent in the radiation monitors in the offgas system and on the main steam lines.

Materials in the Reactor Coolant System are primarily 304 stainless steel and Zircaloy fuel cladding. The reactor water chemistry limits are established to prevent damage to these materials. Limits are placed on chloride concentration and conductivity. The most important limit is that placed on chloride concentration to prevent stress corrosion cracking of the stainless steel. The attached graph, Fig. 4.6-1, illustrates the results of tests on stressed 304 stainless steel specimens. Failures occurred at concentrations above the curve; no failures occurred at concentrations below the curve. According to the data, allowable chloride concentrations could be set several orders of magnitude above the established limit, at the oxygen concentration (0.2-0.3 ppm) experienced during power operation. Zircaloy does not exhibit similar stress corrosion failures.

However, there are various conditions under which the dissolved oxygen content of the reactor coolant water could be higher than 0.2-0.3 ppm, such as refueling, reactor startup, and hot standby. During these periods with steaming rates less

INSERT 6

In the event of a steam line rupture outside the drywell, a more restrictive coolant activity level of 0.2μCi/gm of dose equivalent I-131 was assumed. With this coolant activity level and adverse meteorological conditions, the calculated radiological dose at the site boundary would be less than 30 rem to the thyroid.

4.7 (cont'd)

(4.) See table 4.7-2 for exceptions.

(5.) Acceptance criterion - The combined leakage rate for all penetrations and valves subject to type B and C tests shall be less than 0.60 La. Leakage from containment isolation valves that are sealed with fluid from a seal system may be excluded when determining the combined leakage rate provided that the installed isolation valve seal-water system fluid inventory is sufficient to assure the sealing function for at least 30 days.

d. Other leak rate tests

(1) The leakage rate for containment isolation valves 10-AOV-68A, B (penetration X-13A, B) for Low Pressure Coolant Injection system and 14-AOV-13A, B (penetration X-16A, B) for Core Spray System shall be less than 11 cubic feet per minute per valve (pneumatically tested at 45 psig with ambient temperature) or 10 gallons per minute per valve (hydrostatically) tested at ~~1000~~ 1,035 psig with ambient temperature.

3.7 BASES

A. Primary Containment

The integrity of the primary containment and operation of the Emergency Core Cooling Systems in combination limit the offsite doses to values less than those specified in 10 CFR 100 in the event of a break in the Reactor Coolant System piping. Thus, containment integrity is required whenever the potential for violation of the Reactor Coolant System integrity exists. Concern about such a violation exists whenever the reactor is critical and above atmospheric pressure. An exception to the requirement to maintain primary containment integrity is allowed during core loading and during low power physics testing when ready access to the reactor vessel is required. There will be no pressure on the system at this time, which will greatly reduce the chances of a pipe break. The reactor may be taken critical during this period, however, restrictive operating procedures and operation of the RWM in accordance with Specification 3.3.B.3 minimize the probability of an accident occurring. Procedures in conjunction with the Rod Worth Minimizer Technical Specifications limit individual control worth such that the drop of any in-sequence control rod would not result in a peak fuel enthalpy greater than 280 calories/gm. In the unlikely event that an excursion did occur, the reactor building and Standby Gas Treatment System, which shall be operational during this time, offers a sufficient barrier to keep offsite doses well within 10 CFR 100.

The pressure suppression pool water provides the heat sink for the Reactor Coolant System energy release following a postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat released during reactor coolant system blowdown from ~~1,020~~ 1,040 psig.

Since all of the gases in the drywell are purged into the pressure suppression chamber air space during a loss of coolant accident, the pressure resulting from isothermal compression plus the vapor pressure of the liquid must not exceed 56 psig, the suppression chamber design pressure. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber (Section 5.2).

3.7 BASES (cont'd)

an approximate suppression chamber

Using the minimum or maximum downcomer submergence levels given in the specification, containment pressure during the design basis accident is approximately 45 psig which is below the design of 56 psig. The minimum downcomer submergence of 51.5 in. results in ~~a minimum suppression chamber~~ water volume of 105,600 ft.[3]. The majority of the Bodega tests (9) were run with a submerged length of 4 ft. and with complete condensation. Thus, with respect to downcomer submergence, this specification is adequate. Additional JAFNPP specific analyses done in connection with the Mark I Containment-Suppression Chamber Integrity Program indicate the adequacy of the specified range of submergence to ensure that dynamic forces associated with pool swell do not result in overstress of the suppression chamber or associated structures. Level instrumentation is provided for operator use to maintain downcomer submergence within the specified range.

105,900

The maximum temperature at the end of blowdown tested during the Humboldt Bay (10) and Bodega Bay tests was 170°F, and this is conservatively taken to be ~~the limit for complete condensation of~~ the limit for complete condensation of the reactor coolant, although condensation would occur for temperatures above 170°F.

Insert 7

Using a 40°F rise (Section 5.2 FSAR) in the suppression chamber water temperature and a maximum initial temperature of 95°F, a temperature of 145°F is achieved, which is well below the 170°F temperature which is used for complete condensation.

Insert 8

For an initial maximum suppression chamber water temperature of 95°F and assuming the normal complement of containment cooling pumps (two LPCI pumps and two RHR service water pumps) containment pressure is not required to maintain adequate net positive suction head (HPSH) for the core spray LPCI and HPCI pumps.

Limiting suppression pool temperature to ~~105~~ 95°F during RCIC, HPCI, or relief valve operation, when decay heat and stored energy are removed ~~from~~ the primary system by discharging reactor steam directly to ~~the~~ suppression chamber assures adequate margin for a potential ~~blowdown~~ any time during RCIC, HPCI, or relief valve operation. from

Experimental data indicates that excessive steam condensing loads can be avoided if the peak temperature of the suppression pool is maintained below 160°F during any period of relief valve operation with sonic conditions at the discharge exit. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings.

Insert 9

INSERT 7

Containment analyses predict a 46°F increase in pool water temperature, after complete LOCA blowdown. These analyses assumed an initial suppression pool water temperature of 95°F and a rated reactor power of 2536 MWt. LOCA analyses in Section 14.6 of the FSAR also assume an initial 95°F pool temperature. Therefore, complete condensation is assured during a LOCA because the maximum pool temperature (141°F) is less than the 170°F temperature seen during the Bodega Bay tests.

INSERT 8

For an initial maximum suppression chamber water temperature of 95°F, assuming the worst case complement of containment cooling pumps (one LPCI pump and two RHR service water pumps), containment pressure is required to maintain adequate net positive suction head (NPSH) for the core spray and LPCI pumps.

INSERT 9

Experiments indicate that unacceptably high dynamic containment loads may result from unstable condensation when suppression pool water temperatures are high near SRV discharges. Action statements limit the maximum pool temperature to assure stable condensation. These actions include: limiting the maximum pool temperature of 95°F during normal operation; initiating a reactor scram if during a transient (such as a stuck open SRV) pool temperature exceeds 110°F; and depressurizing the reactor if pool temperature exceeds 120°F. T-quenchers diffuse steam discharged from SRVs and promote stable condensation. The presence of T-quenchers and compliance with these action statements assure that stable condensation will occur and containment loads will be acceptable.

NEDC-24361P (August 1981) summarizes analyses performed to predict pool temperatures and containment loads during plant transients using these temperature limits at a power level of 2535 MWt (104% of rated). NEDC-24361P also substantiates the acceptability of the plant design using the local pool limits of NUREG-0661. NEDO-30832 (December 1984) shows that SRV condensation loads are low compared to other design loads for plants with T-quenchers. NEDO-30832 describes why local pool temperatures need not be analyzed at a rated power level of 2536 MWt.

4.7 BASES

A. Primary Containment

The water in the suppression chamber is used only for cooling in the event of an accident; i.e., it is not used for normal operation; therefore, a daily check of the temperature and volume is adequate to assure that adequate heat removal capability is present.

The primary containment preoperational test pressures are based upon the calculated primary containment pressure response corresponding to the design basis loss-of-coolant accident. The peak drywell pressure would be about 45 psig which would rapidly reduce to 27 psig within 30 sec. following the pipe break. Following the pipe break, the suppression chamber pressure rises to 26 psig within 30 sec, equalizes with drywell pressure and thereafter rapidly decays with the drywell pressure decay (14).

The design pressure of the drywell and suppression chamber is 56 psig(15). The design basis accident leakage rate is 0.5 percent/day at a pressure of 45 psig. As pointed out above, the drywell and suppression chamber pressure following an accident would equalize fairly rapidly. Based on the primary containment pressure response and the fact that the drywell and suppression chamber function as a unit, the primary containment will be tested as a unit rather than the individual components separately.

Insert 10

The design basis loss-of-coolant accident was evaluated in FSAR Section 14.6 incorporating the primary containment maximum allowable accident leak rate of 1.5 percent/day. The analysis showed that with the leak rate and a standby gas treatment system filter efficiency of 99 percent for halogens, 99 percent for particulate and assuming the fission product release fractions stated in TID-14844, the maximum total whole body passing cloud dose is about .97 rem and the maximum total thyroid dose is about 11.4 rem at the site boundary over an exposure duration of two hours. The resultant thyroid dose that would occur over a 30-day period is 32.5 rem at the boundary of the low population zone (LPZ). Thus, these doses are the maximum that would be expected in the unlikely event of a design basis loss-of-coolant accident. These doses are also based on the

Amendment No

INSERT 10

Design basis accidents were evaluated as discussed in Section 14.6 of the FSAR and the power uprate safety evaluation, Reference 18. The whole body and thyroid doses in the control room, low population zone (LPZ) and site boundary meet the requirements of 10 CFR Parts 50 and 100. The technical support center (TSC), not designed to these licensing bases, was also analyzed. The whole body and thyroid dose acceptance criteria used for the main control room are met for the TSC when initial access to the TSC and occupancy of certain areas in the TSC is restricted by administrative control. The LOCA dose evaluation, Reference 19, assumed: the primary containment leak rate was 1.5 volume percent per day; source term releases were in accordance with TID-14844; and the standby gas treatment system filter efficiency was 99% for halogens.

(A) ROUTINE REPORTS (Continued)

4. CORE OPERATING LIMITS REPORT

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:

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

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

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

1. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P, latest approved version and amendments.
2. "James A. FitzPatrick Nuclear Power Plant SAFER/GESTR - LOCA Loss-of-Coolant Accident Analysis," NEDC-31317P, October, 1986 including latest revision, errata and addenda.
3. "Loss-of-Coolant Accident Analysis for James A. FitzPatrick Nuclear Power Plant," NEDO-21662-2, July, 1977 including latest errata and addenda.

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.

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.

7.0 REFERENCES

(1) E. Janssen, "Multi-Rod Burnout at Low Pressure," ASME Paper 62-HT-26, August 1962.

(2) K.M. Backer, "Burnout Conditions for Flow of Boiling Water in Vertical Rod Clusters," AE-74 (Stockholm, Sweden), May 1962.

(3) FSAR Section 11.2.2.

(4) FSAR Section 4.4.3.

(5) I.M. Jacobs, "Reliability of Engineered Safety Features as a Function of Testing Frequency," Nuclear Safety, Vol. 9, No. 4, July-August 1968, pp 310-312.

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(15) ASME Boiler and Pressure Vessel Code, Nuclear Vessels, Section III. Maximum allowable internal pressure is 62 psig.

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(18) General Electric Report NEDC-32016P, "Power Uprate Safety Analysis for the James A. FitzPatrick Nuclear Power Plant," December 1991 (proprietary).

(19) James A. FitzPatrick Calculation JAF-CALC-RAD-00008, "Radiological Consequences of Design Basis Accidents at James A. FitzPatrick," November 1991.

(20) General Electric Report GE-NE-187-45-1191P, "Containment Systems Evaluation," (proprietary).