Attachment I to JPN-96-046

UPDATED PAGE CHANGES FOR PROPOSED CHANGE TO THE TECHNICAL SPECIFICATIONS REGARDING POWER UPRATE (JPTS-91-025)

New York Power Authority

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

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-1, "Power Uprate Safety Analysis for James A. FitzPatrick Nuclear Power Plant," April 1993 (proprietary), Including Errata and Addenda Sheet No. 1, dated January 1994.

1.1 BASES (Cont'd)

C. Power Transient

Plant safety analyses have shown that the scrams caused by exceeding any safety system setting will assure that the Safety Limit of 1.1.A or 1.1.B will not be exceeded. Scram times are checked periodically to assure the insertion times are adequate. The thermal power transient resulting when a scram is accomplished other than by the expected scram signal (e.g., scram from neutron flux following closure of the main turbine stop valves) does not necessarily cause fuel damage. However, for this specification a Safety Limit violation will be assumed when a scram is only accomplished by means of a backup feature of the plant design. The concept of not approaching a Safety Limit provided scram signals are operable is supported by the extensive plant safety analysis.

D. Reactor Water Level (Hot or Cold Shutdown Condition)

During periods when the reactor is shut down, consideration must also be given to water level requirements due to the effect of decay heat. If reactor water level should drop below the top of the active fuel during this time, the ability to cool the core is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and clad perforation. The core will be cooled sufficiently to prevent clad melting should the water level be reduced to two-thirds the core height. Establishment of the Safety Limit at 18 in. above the top of the fuel provides adequate margin. This level will be continuously monitored whenever the recirculation pumps are not operating.

E. References

1. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-13, August 1996.

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

3. GE12 Compliance with Amendment 22 of NEDE-24011-P-A (GESTAR II), NEDE-32417P, December 1994.

2.1 BASES (Cont'd)

B. Not Used

C. <u>References</u>

1. General Electric Report, NEDC-32016P-1, "Power Uprate Safety Analysis for James A. FitzPatrick Nuclear Power Plant," April 1993 (proprietary), Including Errata and Addenda Sheet No. 1, dated January 1994.

2. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-13, August 1996.

3. (Deleted)

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

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 °F 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 limiting vessel overpressure transient event is a main steam isolation valve closure with flux scram. This event was analyzed within NEDC-32016P-1, "Power Uprate Safety Analysis For James A. FitzPatrick Nuclear Power Plant," including Errata and Addenda Sheet No. 1, dated January 1994, assuming 9 of the 11 SRVs were operable with opening pressures less than or equal to 1179 psig. The resultant peak vessel pressure for the event was shown to be less than the ASME Code limit of 1375 psig (see current reload analysis for the reactor response to the main steam isolation valve closure with flux scram event).

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.

3.1 BASES (cont'd)

Turbine control valves fast closures 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 the MCPR value for each fuel type as described in Reference 1. 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.

C. References

1. "James A. FitzPatrick Nuclear Power Plant SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," NEDC-31317P, Revision 2, April 1993.

3.3 and 4.3 BASES (cont'd)

"full out" position during the performance of SR 4.3.A.2.a. This Frequency is acceptable, considering the low probability that a control rod will become uncoupled when it is not being moved, and operating experience related to uncoupling events.

2. The control rod housing support restricts the outward movement of a control rod to less than 3 in. in the extremely remote event of a housing failure. The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage to the Primary Coolant System. The design basis is given in subsection 3.8.2 of the FSAR, and the safety evaluation is given in subsection 3.8.4. This support is not required if the Reactor Coolant System is at atmospheric pressure since there would then be no driving force to rapidly eject a drive housing. Additionally, the support is not required if all control rods are fully inserted and if an adequate shutdown margin with one control rod withdrawn has been demonstrated, since the reactor would remain subcritical even in the event of complete ejection of the strongest control rod.

3. The Rod Worth Minimizer (RWM) restricts the order of control rod withdrawal and insertion to be equivalent to the Banked Position Withdrawal Sequence (BPWS). These sequences are established such that the drop of any in-sequence control rod from the fully inserted position to the position of the control rod drive would not cause the reactor to sustain a power excursion resulting in a peak fuel enthalpy in excess of 280 cal/gm. An enthalpy of 280 cal/gm is well below the level at which rapid fuel dispersal could occur (i.e. 425 cal/gm.). Primary system damage in this accident is not possible unless a significant amount of fuel is rapidly dispersed. Ref. Subsections 3.6.6, 7.7.4.3 and 14.6.1.2 of the FSAR, NEDE-24011-P-A-13, August 1996 and NEDO-10527 including supplements 1 and 2 to NEDO-10527.

In performing the function described above, the RWM is not required to impose any restrictions at core power levels in excess of 10% of rated. Material in the cited references shows that it is impossible to reach 280 calories per gram in the event of a control rod drop occurring at power greater than 10%, regardless of the rod pattern. This is true for all normal and abnormal patterns including those which maximize the individual control rod worth.

3.3 and 4.3 **BASES** (cont'd)

5. The Rod Block Monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power level operation. Two channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Tripping of one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage.

   This system backs up the operator who withdraws control rods according to written sequences. The specified restrictions with one channel out of service conservatively assure that fuel damage will not occur due to rod withdrawal errors when this condition exists.

   A limiting control rod pattern is a pattern which results in the core being on a thermal hydraulic limit (e.g., MCPR limit). During use of such patterns, it is judged that testing of the RBM System prior to withdrawal of such rods to assure its operability will assure that improper withdraw does not occur. It is the responsibility of the Reactor Engineer to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable control rods in other than limiting patterns.

C. Scram Insertion Times

The Control Rod System is designated to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCPR from becoming less than the Safety Limit. Scram insertion time test criteria of Section 3.3.C.1 were used to generate the generic scram reactivity curve shown in NEDE-24011-P-A-13, August 1996. This generic curve was used in analysis of non-pressurization transients to determine MCPR limits. Therefore, the required protection is provided.

The numerical values assigned to the specified scram performance are based on the analysis of data from other BWR's with control rod drives the same as those on JAFNPP.

The occurrence of scram times within the limits, but significantly longer than the average, should be viewed as an indication of a systematic problem with control rod drives, especially if the number of drives exhibiting such scram times exceeds eight, the allowable number of inoperable rods.

3.5 BASES

A. Core Spray System and Low Pressure Coolant injection (LPCI) Mode of the RHR System

This specification assures that adequate emergency cooling capability is available whenever irradiated fuel is in the reactor vessel.

The loss-of-coolant analysis is referenced and described in General Electric Topical Report NEDE-24011-P-A-13, August 1996.

The limiting conditions of operation in Specifications 3.5.A.1 through 3.5.A.6 specify the combinations of operable subsystems to assure the availability of the minimum cooling systems. No single failure of ECCS equipment occurring during a loss-of-coolant accident under these limiting conditions of operation will result in inadequate cooling of the reactor core.

Core spray distribution has been shown, in full scale tests of systems similar in design to that of the FitzPatrick Plant, to exceed the minimum requirements by at least 25 percent. In addition, cooling effectiveness has been demonstrated at less than half the rated flow in simulated fuel assemblies with heater rods to duplicate the decay heat characteristics of irradiated fuel. The accident analysis is additionally conservative in that no credit is taken for spray coolant entering the reactor before the internal pressure has fallen to 113 psi above primary containment pressure.

The LPCI mode of the RHR System is designed to provide emergency cooling to the core by flooding in the event of a loss-of-coolant accident. These subsystems are completely independent of the Core Spray System; however, they function in combination with the Core Spray System to prevent excessive fuel clad temperature. The LPCI mode of

3.6 and 4.6 BASES (cont'd)

E. Safety/Relief Valves

The safety/relief valves (SRVs) have two modes of operation; the safety mode or the relief mode. In the safety mode (or spring mode of operation) the spring loaded pilot valve opens when the steam pressure at the valve inlet overcomes the spring force holding the pilot valve closed. The safety mode of operation is required during pressurization transients to ensure vessel pressures do not exceed the reactor coolant pressure safety limit of 1,375 psig.

In the relief mode the spring loaded pilot valve opens when the spring force is overcome by nitrogen pressure which is provided to the valve through a solenoid operated valve. The solenoid operated valve is actuated by the ADS logic system (for those SRVs which are included in the ADS) or manually by the operator from a control switch in the main control room or at the remote ADS panel. Operation of the SRVs in the relief mode for the ADS is discussed in the Bases for Specification 3.5.D.

Experiences in safety/relief valve testing have shown that failure or deterioration of safety/relief valves can be adequately detected if at least 5 of the 11 valves are bench tested once every 24 months so that all valves are tested every 48 months. Furthermore, safety/relief valve testing experience has demonstrated that safety/relief valves which actuate within $\pm 3\%$ of the design pressure setpoint are considered operable (see ANSI/ASME OM-1-1981). The safety bases for a single nominal valve opening pressure of 1145 psig are described in NEDC-32016P-1, "Power Uprate Safety Analysis for James A. FitzPatrick Nuclear Power Plant," Including Errata and Addenda Sheet No. 1, dated January 1994. The single nominal setpoint is set below the reactor vessel design pressure (1250 psig) per the requirements of Article 9 of the ASME Code - Section III, Nuclear Vessels. The setting of 1145 psig preserves the safety margins associated with the HPCI and RCIC turbine overspeed systems and the Mark I torus loading analyses. Based on safety/relief valve testing experience and the analysis referenced above, the safety/relief valves are bench tested to demonstrate that in-service opening pressures are within the nominal pressure setpoints $\pm 3\%$ and then the valves are returned to service with opening pressures at the nominal setpoints $\pm 1\%$. In this manner, valve integrity is maintained from cycle to cycle.

The analyses with NEDC-32016P-1, Including Errata and Addenda Sheet No. 1, dated January 1994, also provide the safety basis for which 2 SRVs are permitted inoperable during continuous power operation. With more than 2 SRVs inoperable, the margin to the reactor vessel pressure safety limit is significantly reduced, therefore, the plant must enter a cold condition within 24 hours once more than 2 SRVs are determined to be inoperable. (See reload evaluation for the current cycle).

A manual actuation of each SRV is performed to demonstrate that the valves are mechanically functional and that no blockage exists in the valve discharge line. Valve opening is confirmed by monitoring the response of the turbine bypass valves and the SRV acoustic monitors. Adequate reactor steam dome pressure must be available to avoid damaging the valve. Adequate steam flow is required to ensure that reactor pressure can be maintained during the test. Testing is performed in the RUN mode to reduce the risk of a reactor scram in response to small pressure fluctuations which may occur while opening and reclosing the valves.

Low power physics testing and reactor operator training with inoperable components will be conducted only when the safety/relief valves are

3.7 (cont'd)

(2) During testing which adds heat to the suppression pool, the water temperature shall not exceed 10°F above the normal power operation limit specified in (1) above. In connection with such testing, the pool temperature must be reduced to below the normal power operation limit specified in (1) above within 24 hours.

(3) The reactor shall be scrammed from any operating condition if the pool temperature reaches 110°F. Power operation shall not be resumed until the pool temperature is reduced below the normal power operation limit specified in (1) above.

(4) During reactor isolation conditions, the reactor pressure vessel shall be depressurized to less than 200 psig at normal cooldown rates if the pool temperature reaches 120°F.

2. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F, and fuel is in the reactor vessel, except while performing low power physics tests at atmospheric pressure at power levels not to exceed 5 MWt.

4.7 (cont'd)

2. a. Perform required visual examination and leakage rate testing of the Primary Containment in accordance with the Primary Containment Leakage Rate Testing Program.

b. Demonstrate leakage rate through each MSIV is $\leq$ 11.5 scfh when tested at $\geq$ 25 psig. The testing frequency is in accordance with the Primary Containment Leakage Rate Testing Program.

c. Once per 24 months, demonstrate the leakage rate of 10AOV-68A,B for the Low Pressure Coolant Injection system and 14AOV-13A,B for the Core Spray system to be less than 11 scfm per valve when pneumatically tested at $\geq$ 45 psig at ambient temperature, or less than 10 gpm per valve if hydrostatically tested at $\geq$ 1,035 psig at ambient temperature.

3.7 BASES (cont'd)

Using the minimum or maximum torus water level (which are based on downcomer submergence levels where 13.88 feet above the bottom of the torus is 0.005 feet higher than the minimum submergence of 51.5 inches and 14.00 feet above the bottom of the torus is equivalent to the maximum submergence of 53 inches assumed in containment analyses) 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 inches results in a minimum torus water volume of approximately 105,900 feet^3. The majority of the Bodega tests (9) were run with a submerged length of 4 feet 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 torus 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 torus 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 torus 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.

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 evaluations, References 19, 20, and 21, assumed: the primary containment leak rate was 1.5 volume percent per day; source term releases were in accordance with TID-14844 and Regulatory Guide 1.3, and were consistent with the Standard Review Plan; and the standby gas treatment system filter efficiency was 99% for halogens. These doses are also based on the

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) Deleted

(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) Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program", dated September 1995.

(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) 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors, Option B - Performance Based Requirements", Effective Date October 26, 1995

(17) Deleted

(18) General Electric Report NEDC-32016P-1, "Power Uprate Safety Analysis for James A. FitzPatrick Nuclear Power Plant," April 1993 (proprietary), Including Errata and Addenda Sheet No. 1, dated January 1994.

(19) James A. FitzPatrick Calculation JAF-CALC-RAD-00023, Rev. 0, "Power Uprate Program - Technical Support Center Post-Accident Radiological Habitability Study," August 1996.

(20) James A. FitzPatrick Calculation JAF-CALC-RAD-00042, Rev. 0, "Control Room Radiological Habitability Under Power Uprate Conditions and CREVASS Reconfiguration," September 1995.

7.0 <u>REFERENCES</u> (continued)

(21) James A. FitzPatrick Calculation JAF-CALC-RAD-00048, Rev. 0, "Power Uprate Project - Radiological Impact at Onsite and Offsite Outdoor Receptors Following Design-Basis Accidents," May 1996

(22) General Electric Report GE-NE-187-45-1191, "Containment Systems Evaluation for the James A. FitzPatrick Nuclear Power Plant," November 1991 (proprietary).

Attachment II to JPN-96-046

EVALUATION OF UPDATED PAGE CHANGES FOR PROPOSED CHANGE TO THE TECHNICAL SPECIFICATIONS REGARDING POWER UPRATE (JPTS-91-025)

New York Power Authority

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

I. DESCRIPTION OF THE PROPOSED CHANGES

This section provides a description of the proposed changes to the Technical Specifications (TS). Minor changes in format, such as type font, margins or hyphenation, are not described in this submittal. These changes are typographical and do not affect the content of the TS.

Technical Specifications Page 6a, Definitions Section 1.0.AE.1

This change adds the current revision of the Power Uprate Safety Analysis Report (NEDC-32016P-1, Reference 1). This change supersedes the change originally proposed in JPN-92-028 (Reference 2):

Add the following:

"AE. References

1. General Electric Report NEDC-32016P-1, 'Power Uprate Safety Analysis for James A. FitzPatrick Nuclear Power Plant,' April 1993 (proprietary), Including Errata and Addenda Sheet No. 1, dated January 1994"

Technical Specifications Page 14, Bases Section 1.1.E.1

This change adds the current revision of the General Electric Standard Application for Reactor Fuel. This change is necessary to reflect the change proposed in JPN-96-043 (Reference 3) to TS Page 254c, Section 6.9.(A).4.b.1.

Replace the following:

"1. 'General Electric Standard Application for Reactor Fuel,' NEDE-24011-P, latest approved revision and amendments."

with:

"1. 'General Electric Standard Application for Reactor Fuel,' NEDE-24011-P-A-13, August 1996"

Technical Specifications Page 20, Bases Section 2.1.C.1

This change adds the current revision of the Power Uprate Safety Analysis Report (NEDC-32016P-1, Reference 1). The change supersedes the change originally proposed in JPN-92-028 (Reference 2) and in JAFP-96-0306 (Reference 4).

Replace:

"1. (Deleted)"

with:

"1. General Electric Report, NEDC-32016P-1, 'Power Uprate Safety Analysis for James A. FitzPatrick Nuclear Power Plant,' April 1993 (proprietary), Including Errata and Addenda Sheet No. 1, dated January 1994."

Technical Specifications Page 20, Bases Section 2.1.C.2

This change adds the current revision of the General Electric Standard Application for Reactor Fuel. This change is necessary to reflect the change proposed in JPN-96-043 (Reference 3) to TS Page 254c, Section 6.9.(A).4.b.1.

Replace the following:

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

with:

"2. 'General Electric Standard Application for Reactor Fuel,' NEDE-24011-P-A-13, August 1996."

Technical Specifications Page 29, Bases Section 1.2 and 2.2

This change adds the current revision of the Power Uprate Safety Analysis Report (NEDC-32016P-1, Reference 1).

Replace the following change originally proposed in JPN-92-028 (Reference 2) and superseded by the change proposed in JAFP-96-0306 (Reference 4):

"...NEDC-31697P, 'Updated SRV Performance Requirements for the JAFNPP,' assuming 9 of the 11 SRVs were operable with opening pressures less than or equal to 1195 psig. The resultant peak vessel pressure for the event was shown to be less than the vessel pressure code limit of 1375 psig. (See current reload analysis for the reactor response to the main steam isolation valve closure with flux scram event). The value of 1195 psig is the SRV opening pressure up to which plant performance has been analyzed, assuming 2 SRVs are inoperable. Therefore, SRV opening pressures below 1195 psig ensure that the ASME Code limit on peak reactor pressure is satisfied..."

with:

"...NEDC-32016P-1, 'Power Uprate Safety Analysis For James A. FitzPatrick Nuclear Power Plant,' Including Errata and Addenda Sheet No. 1, dated January 1994, assuming 9 of the 11 SRVs were operable with opening pressures less than or equal to 1179 psig. The resultant peak vessel pressure for the event was shown to be less than the ASME Code limit of 1375 psig (see current reload analysis for the reactor response to the main steam isolation valve closure with flux scram event)..."

Technical Specifications Page 35, Bases Sections 3.1.B, 3.1.C.1, and 3.1.C.2

This change deletes reference to an outdated Loss of Coolant Accident Analysis Report for FitzPatrick (NEDO-21662, Reference 5). In addition, this change provides the current reference for the Loss of Coolant Accident Analysis Report applicable to FitzPatrick (NEDC-31317P, Reference 6). These changes are necessary to reflect the changes proposed in JPN-96-043 (Reference 3) to TS Page 254c, Section 6.9.(A).4.b. JAFP-96-0306 (Reference 4) stated that no change was required to TS Page 35. However, for the reasons stated above, a change is required.

1. Replace the following in Bases Section 3.1.B:

   "...NEDO-21662 (Reference 1) and NEDC-31317P (Reference 2) including the latest revision, errata and addenda..."

   with:

   "...Reference 1..."

2. Replace the following from Bases Section 3.1.C:

   "1. General Electric Topical Report NEDO-21662, Revision 2, 'Loss-of-Coolant Accident Analysis Report for James A. FitzPatrick Nuclear Power Plant (Lead Plant)', July 1977 with errata and addenda.

   2. General Electric Topical Report NEDC-31317P, 'James A. FitzPatrick Nuclear Power Plant SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis', October 1986 with revisions, errata and addenda."

   with:

   "1. 'James A. FitzPatrick Nuclear Power Plant SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis,' NEDC-31317P, Revision 2, April 1993."

Technical Specifications Page 100, Section 3.3 and 4.3 of the Bases Item B.3

This change adds the current revision of the General Electric Standard Application for Reactor Fuel. This change is necessary to reflect the change proposed in JPN-96-043 (Reference 3) to TS Page 254c, Section 6.9.(A).4.b.1.

Replace the following:

"...NEDE-24011..."

with:

"...NEDE-24011-P-A-13, August 1996..."

<u>Technical Specifications Page 102, Section 3.3 and 4.3 of the Bases Item C.</u>

This change adds the current revision of the General Electric Standard Application for Reactor Fuel. This change is necessary to reflect the change proposed in JPN-96-043 (Reference 3) to TS Page 254c, Section 6.9.(A).4.b.1.

Replace the following:

"...NEDE-24011-P-A..."

with:

"...NEDE-24011-P-A-13, August 1996..."

<u>Technical Specifications Page 125, Bases Section 3.5.A.</u>

This change adds the current revision of the General Electric Standard Application for Reactor Fuel. This change is necessary to reflect the change proposed in JPN-96-043 (Reference 3) to TS Page 254c, Section 6.9.(A).4.b.1.

Replace the following:

"...NEDE-24011-P-A..."

with:

"...NEDE-24011-P-A-13, August 1996..."

<u>Technical Specifications Page 152, Section 3.6 and 4.6 of the Bases Item E.</u>

This change adds the current revision of the Power Uprate Safety Analysis Report (NEDC-32016P-1, Reference 1) and adds a revision bar which was omitted in JAFP-96-0306. This change supersedes the change originally proposed in JPN-92-028 (Reference 2) and in JAFP-96-0306 (Reference 4).

Replace:

"...The safety bases for a single nominal valve opening pressure of 1110 psig are described in NEDC-31697P, 'Updated SRV Performance Requirements for the JAFNPP.' The single nominal setpoint is set below the reactor vessel design pressure (1250 psig) per the requirements of Article 9 of the ASME Code - Section III, Nuclear Vessels. The setting of 1110 psig..."

and:

"...The analyses with NEDC-31697P also..."

with

"The safety bases for a single nominal valve opening pressure of 1145 psig are described in NEDC-32016P-1, 'Power Uprate Safety Analysis for James A. FitzPatrick Nuclear Power Plant,' Including Errata and Addenda Sheet No. 1, dated January 1994. The single nominal setpoint is set below the reactor vessel design pressure (1250 psig) per the requirements of Article 9 of the ASME Code - Section III, Nuclear Vessels. The setting of 1145 psig..."

and:

"...The analyses with NEDC-32016P-1, Including Errata and Addenda Sheet No. 1, dated January 1994, also..."

<u>Technical Specifications Page 166, Section 4.7.A.2.c.</u>

This change is made to reflect the issuance of TS Amendment 234 (Reference 7). Amendment 234, in part, deleted TS Page 172, and relocated TS Section 4.7.A.2.d.(1) to TS Page 166 and renumbered this section as 4.7.A.2.c. This change affects relocation only and does not affect the technical information previously submitted under JPN-92-028 (Reference 2).

Replace:

"...hydrostatically tested at ≥ 1000 psig..."

with:

"...hydrostatically tested at ≥ 1,035 psig..."

<u>Technical Specifications Page 172, Section 4.7.A.2.d.(1)</u>

No change is required to TS Page 172. Amendment 234, in part, deleted TS Page 172.

<u>Technical Specifications Page 188, BASES Section 3.7</u>

This change only places certain revision bars on TS page 188 in the correct position. Revision bars on this page were not correctly shown in JAFP-96-0306 (Reference 4).

<u>Technical Specifications Page 193, BASES Section 4.7</u>

Add References 20 and 21 after 19 in the proposed submittal (i.e., JPN-92-028). These three references (i.e., 19, 20, and 21) reflect the current calculations regarding radiological consequences of design basis accidents.

Add to the proposed submittal (JPN-92-028) that the LOCA dose evaluations, References 19, 20, and 21, assumed source term releases were in accordance with TID-14844 and Regulatory Guide 1.3, and were consistent with the Standard Review Plan (NUREG-0800).

These changes do not alter and are consistent with the conclusions presented in References 1 and 2. The Authority previously submitted updated radiological consequences of design basis accidents to the NRC under Reference 4.

These are the only changes to that previously submitted under JPN-92-028.

Replace:

"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."

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 evaluations, References 19, 20, and 21, assumed: the primary containment leak rate was 1.5 volume percent per day; source term releases were in accordance with TID-14844 and Regulatory Guide 1.3, and were consistent with the Standard Review Plan; and the standby gas treatment system filter efficiency was 99% for halogens."

<u>Technical Specifications Page 285 and 285a, References Section 7.0</u>

New Page 285a is added to the TS to reflect additional references.

There is no change to Reference 10 from that previously submitted under JAFP-96-0426 (Reference 8). JAFP-96-0426 superseded JPN-92-028 (Reference 2) and JAFP-96-0306 (Reference 4) with regards to Page 285.

Reference 18 is changed to add the current revision of the Power Uprate Safety Analysis Report (NEDC-32016P-1, Reference 1) from that previously submitted under JAFP-96-0426.

Proposed Reference 19 is deleted and replaced with three references, Numbered 19, 20, and 21. These three references reflect the current calculations regarding radiological consequences of design basis accidents. Reference 21 will be located on Page 285a. This change supersedes the Reference 19 change submitted under JAFP-96-0426.

Proposed Reference 20 is renumbered as Reference 22 (GE-NE-187-45-1191) due to addition of references regarding radiological consequences. In addition, JAFP-96-0426 erroneously stated that a "P" was at the end of this document. (i.e., GE-NE-187-45-1191P). Although this document is proprietary, there is no "P" at the end of the document number. As such, the "P" has been deleted. This reference will be located on TS Page 285a. This change supersedes the Reference 20 change submitted under JAFP-96-0426.

1. Replace the following on Page 285:

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

   with:

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

2. add the following to Page 285:

   "(18) General Electric Report NEDC-32016P-1, 'Power Uprate Safety Analysis for James A. FitzPatrick Nuclear Power Plant.' April 1993 (proprietary), Including Errata and Addenda Sheet No. 1, dated January 1994.

   (19) James A. FitzPatrick Calculation JAF-CALC-RAD-00023, Rev. 0, 'Power Uprate Program - Technical Support Center Post-Accident Radiological Habitability Study,' August 1996.

   (20) James A. FitzPatrick Calculation JAF-CALC-RAD-00042, Rev. 0, 'Control Room Radiological Habitability Under Power Uprate Conditions and CREVASS Reconfiguration,' September 1995."

3. add the following to Page 285a:

   "(21) James A. FitzPatrick Calculation JAF-CALC-RAD-00048, Rev. 0, 'Power Uprate Project - Radiological Impact at Onsite and Offsite Outdoor Receptors Following Design-Basis Accidents,' May 1996

   (22) General Electric Report GE-NE-187-45-1191, "Containment Systems Evaluation for the James A. FitzPatrick Nuclear Power Plant," November 1991."

II. SAFETY IMPLICATIONS OF THE PROPOSED CHANGES

There are no safety implications associated with these proposed changes. The Authority has reviewed these proposed changes and has determined that adoption of these changes do not affect the bases or conclusions of the no significant hazards considerations described in NEDC-32016P-1 (Reference 1) and in JPN-92-028 (Reference 2). The TS changes provided in Attachment I are administrative in nature and support the overall conclusions presented in References 1 and 2. The PORC and SRC have reviewed these proposed changes to the TS and concur with this conclusion.

VII. REFERENCES

1. General Electric Report NEDC-32016P-1, "Power Uprate Safety Analysis for James A. FitzPatrick Nuclear Power Plant," April 1993 (proprietary), Including Errata and Addenda Sheet No. 1, dated January 1994

2. NYPA Letter, R. E. Beedle to the NRC, (JPN-92-028), "Proposed Changes to the Technical Specifications Regarding Power Uprate (JPTS-91-025)," dated June 12, 1992

3. NYPA Letter, W. J. Cahill, Jr. to the NRC, (JPN-96-043), "Response to Request for Additional Information Regarding Power Uprate," dated November 14, 1996

4. NYPA Letter, M. J. Colomb to the NRC, (JAFP-96-0306), "Updated Page Changes for Proposed Change to the Technical Specifications Regarding Power Uprate (JPTS-91-025)," dated August 15, 1996

5. General Electric Topical Report NEDO-21662, Revision 2, "Loss-of-Coolant Accident Analysis Report for James A. FitzPatrick Nuclear Power Plant (Lead Plant)," July 1977 with errata and addenda.

6. General Electric Topical Report NEDC-31317P, "James A. FitzPatrick Nuclear Power Plant SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," October 1986 with revisions, errata and addenda

7. NRC Letter, K. R. Cotton to W. J. Cahill Jr., Regarding Issuance of Amendment 234 for the James A. FitzPatrick Nuclear Power Plant (TAC No. M95099), dated October 4, 1996

8. NYPA Letter, M. J. Colomb to the NRC, (JAFP-96-0426), "Submittal of Updated Pages Regarding Proposed Changes to the Technical Specifications Contained in the Referenced Letters," dated October 23, 1996

Attachment III to JPN-96-046

MARKUP TO REFLECT UPDATED PAGE CHANGES FOR PROPOSED CHANGE TO THE TECHNICAL SPECIFICATIONS REGARDING POWER UPRATE (JPTS-91-025)

NOTE 1: Deletions are shown in ~~strikeout~~, and additions are in **bold**.

NOTE 2: Previous amendment revision bars are shown and will be deleted.

New York Power Authority

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

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-1, "Power Uprate Safety Analysis for James A. FitzPatrick Nuclear Power Plant," April 1993 (proprietary), Including Errata and Addenda Sheet No. 1, dated January 1994.

1.1 BASES (Cont'd)

C. Power Transient

Plant safety analyses have shown that the scrams caused by exceeding any safety system setting will assure that the Safety Limit of 1.1.A or 1.1.B will not be exceeded. Scram times are checked periodically to assure the insertion times are adequate. The thermal power transient resulting when a scram is accomplished other than by the expected scram signal (e.g., scram from neutron flux following closure of the main turbine stop valves) does not necessarily cause fuel damage. However, for this specification a Safety Limit violation will be assumed when a scram is only accomplished by means of a backup feature of the plant design. The concept of not approaching a Safety Limit provided scram signals are operable is supported by the extensive plant safety analysis.

D. Reactor Water Level (Hot or Cold Shutdown Condition)

During periods when the reactor is shut down, consideration must also be given to water level requirements due to the effect of decay heat. If reactor water level should drop below the top of the active fuel during this time, the ability to cool the core is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and clad perforation. The core will be cooled sufficiently to prevent clad melting should the water level be reduced to two-thirds the core height. Establishment of the Safety Limit at 18 in. above the top of the fuel provides adequate margin. This level will be continuously monitored whenever the recirculation pumps are not operating.

E. References

1. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P~~, latest approved revision and amendments.~~ **-A-13, August 1996**.

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

3. GE12 Compliance with Amendment 22 of NEDE-24011-P-A (GESTAR II), NEDE-32417P, December 1994.

2.1 BASES (Cont'd)

B. Not Used

C. References

1. ~~(Deleted)~~ **General Electric Report, NEDC-32016P-1, "Power Uprate Safety Analysis for James A. FitzPatrick Nuclear Power Plant," April 1993 (proprietary), Including Errata and Addenda Sheet No. 1, dated January 1994.**

2. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A ~~(Approved revision number applicable at time that reload fuel analyses are performed)~~ **13, August 1996**.

3. (Deleted)

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

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 °F 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 limiting vessel overpressure transient event is a main steam isolation valve closure with flux scram. This event was analyzed within ~~NEDC 31697P, "Updated SRV Performance Requirements for the JAFNPP," assuming 9 of the 11 SRVs were operable with opening pressures less than or equal to 1195 psig. The resultant peak vessel pressure for the event was shown to be less than the vessel pressure code limit of 1375 psig. (See current reload analysis for the reactor response to the main steam isolation valve closure with flux scram event). The value of 1195 psig is the SRV opening pressure up to which plant performance has been analyzed, assuming 2 SRVs are inoperable. Therefore, SRV opening pressures below 1195 psig ensure that the ASME Code limit on peak reactor pressure is satisfied.~~**NEDC-32016P-1, "Power Uprate Safety Analysis For James A. FitzPatrick Nuclear Power Plant," Including Errata and Addenda Sheet No. 1, dated January 1994, assuming 9 of the 11 SRVs were operable with opening pressures less than or equal to 1179 psig. The resultant peak vessel pressure for the event was shown to be less than the ASME Code limit of 1375 psig (see current reload analysis for the reactor response to the main steam isolation valve closure with flux scram event).**

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.

3.1 BASES (cont'd)

Turbine control valves fast closures 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 the MCPR value for each fuel type as described in ~~NEDO-21662 (Reference 1) and NEDC-31317P (Reference 2) including the latest revision, errata and addenda~~ **Reference 1**. 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.

C. References

1. ~~General Electric Topical Report NEDO-21662, Revision 2, "Loss of Coolant Accident Analysis Report for James A. FitzPatrick Nuclear Power Plant (Lead Plant)", July 1977 with errata and addenda.~~

~~2. General Electric Topical Report NEDC-31317P,~~ "James A. FitzPatrick Nuclear Power Plant SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis,"~~, October 1986 with revisions, errata and addenda~~ **NEDC-31317P, Revision 2, April 1993**.

3.3 and 4.3 BASES (cont'd)

"full out" position during the performance of SR 4.3.A.2.a. This Frequency is acceptable, considering the low probability that a control rod will become uncoupled when it is not being moved, and operating experience related to uncoupling events.

2. The control rod housing support restricts the outward movement of a control rod to less than 3 in. in the extremely remote event of a housing failure. The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage to the Primary Coolant System. The design basis is given in subsection 3.8.2 of the FSAR, and the safety evaluation is given in subsection 3.8.4. This support is not required if the Reactor Coolant System is at atmospheric pressure since there would then be no driving force to rapidly eject a drive housing. Additionally, the support is not required if all control rods are fully inserted and if an adequate shutdown margin with one control rod withdrawn has been demonstrated, since the reactor would remain subcritical even in the event of complete ejection of the strongest control rod.

3. The Rod Worth Minimizer (RWM) restricts the order of control rod withdrawal and insertion to be equivalent to the Banked Position Withdrawal Sequence (BPWS). These sequences are established such that the drop of any in-sequence control rod from the fully inserted position to the position of the control rod drive would not cause the reactor to sustain a power excursion resulting in a peak fuel enthalpy in excess of 280 cal/gm. An enthalpy of 280 cal/gm is well below the level at which rapid fuel dispersal could occur (i.e. 425 cal/gm.). Primary system damage in this accident is not possible unless a significant amount of fuel is rapidly dispersed. Ref. Subsections 3.6.6, 7.7.4.3 and 14.6.1.2 of the FSAR, NEDE-24011-**P-A-13, August 1996** and NEDO-10527 including supplements 1 and 2 to NEDO-10527.

In performing the function described above, the RWM is not required to impose any restrictions at core power levels in excess of 10% of rated. Material in the cited references shows that it is impossible to reach 280 calories per gram in the event of a control rod drop occurring at power greater than 10%, regardless of the rod pattern. This is true for all normal and abnormal patterns including those which maximize the individual control rod worth.

3.3 and 4.3 BASES (cont'd)

5. The Rod Block Monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power level operation. Two channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Tripping of one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage.

This system backs up the operator who withdraws control rods according to written sequences. The specified restrictions with one channel out of service conservatively assure that fuel damage will not occur due to rod withdrawal errors when this condition exists.

A limiting control rod pattern is a pattern which results in the core being on a thermal hydraulic limit (e.g., MCPR limit). During use of such patterns, it is judged that testing of the RBM System prior to withdrawal of such rods to assure its operability will assure that improper withdraw does not occur. It is the responsibility of the Reactor Engineer to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable control rods in other than limiting patterns.

C. Scram Insertion Times

The Control Rod System is designated to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCPR from becoming less than the Safety Limit. Scram insertion time test criteria of Section 3.3.C.1 were used to generate the generic scram reactivity curve shown in NEDE-24011-P-A-13, August 1996. This generic curve was used in analysis of non-pressurization transients to determine MCPR limits. Therefore, the required protection is provided.

The numerical values assigned to the specified scram performance are based on the analysis of data from other BWR's with control rod drives the same as those on JAFNPP.

The occurrence of scram times within the limits, but significantly longer than the average, should be viewed as an indication of a systematic problem with control rod drives, especially if the number of drives exhibiting such scram times exceeds eight, the allowable number of inoperable rods.

3.5 BASES

A. Core Spray System and Low Pressure Coolant injection (LPCI) Mode of the RHR System

This specification assures that adequate emergency cooling capability is available whenever irradiated fuel is in the reactor vessel.

The loss-of-coolant analysis is referenced and described in General Electric Topical Report NEDE-24011-P-A-**13, August 1996**.

The limiting conditions of operation in Specifications 3.5.A.1 through 3.5.A.6 specify the combinations of operable subsystems to assure the availability of the minimum cooling systems. No single failure of ECCS equipment occurring during a loss-of-coolant accident under these limiting conditions of operation will result in inadequate cooling of the reactor core.

Core spray distribution has been shown, in full scale tests of systems similar in design to that of the FitzPatrick Plant, to exceed the minimum requirements by at least 25 percent. In addition, cooling effectiveness has been demonstrated at less than half the rated flow in simulated fuel assemblies with heater rods to duplicate the decay heat characteristics of irradiated fuel. The accident analysis is additionally conservative in that no credit is taken for spray coolant entering the reactor before the internal pressure has fallen to 113 psi above primary containment pressure.

The LPCI mode of the RHR System is designed to provide emergency cooling to the core by flooding in the event of a loss-of-coolant accident. These subsystems are completely independent of the Core Spray System; however, they function in combination with the Core Spray System to prevent excessive fuel clad temperature. The LPCI mode of

3.6 and 4.6 BASES (cont'd)

E. Safety/Relief Valves

The safety/relief valves (SRVs) have two modes of operation; the safety mode or the relief mode. In the safety mode (or spring mode of operation) the spring loaded pilot valve opens when the steam pressure at the valve inlet overcomes the spring force holding the pilot valve closed. The safety mode of operation is required during pressurization transients to ensure vessel pressures do not exceed the reactor coolant pressure safety limit of 1,375 psig.

In the relief mode the spring loaded pilot valve opens when the spring force is overcome by nitrogen pressure which is provided to the valve through a solenoid operated valve. The solenoid operated valve is actuated by the ADS logic system (for those SRVs which are included in the ADS) or manually by the operator from a control switch in the main control room or at the remote ADS panel. Operation of the SRVs in the relief mode for the ADS is discussed in the Bases for Specification 3.5.D.

Experiences in safety/relief valve testing have shown that failure or deterioration of safety/relief valves can be adequately detected if at least 5 of the 11 valves are bench tested once every 24 months so that all valves are tested every 48 months. Furthermore, safety/relief valve testing experience has demonstrated that safety/relief valves which actuate within $\pm$3% of the design pressure setpoint are considered operable (see ANSI/ASME OM-1-1981). The safety bases for a single nominal valve opening pressure of ~~1110~~ **1145** psig are described in NEDC-~~31697P, "Updated SRV Performance Requirements for the JAFNPP."~~ **32016P-1, "Power Uprate Safety Analysis for James A. FitzPatrick Nuclear Power Plant," Including Errata and Addenda Sheet No. 1, dated January 1994.** The single nominal setpoint is set below the reactor vessel design pressure (1250 psig) per the requirements of Article 9 of the ASME Code - Section III, Nuclear Vessels. The setting of ~~1110~~ **1145** psig preserves the safety margins associated with the HPCI and RCIC turbine overspeed systems and the Mark I torus loading analyses. Based on safety/relief valve testing experience and the analysis referenced above, the safety/relief valves are bench tested to demonstrate that in-service opening pressures are within the nominal pressure setpoints $\pm$3% and then the valves are returned to service with opening pressures at the nominal setpoints $\pm$1%. In this manner, valve integrity is maintained from cycle to cycle.

The analyses with NEDC-~~31697P~~ **32016P-1, Including Errata and Addenda Sheet No. 1, dated January 1994,** also provide the safety basis for which 2 SRVs are permitted inoperable during continuous power operation. With more than 2 SRVs inoperable, the margin to the reactor vessel pressure safety limit is significantly reduced, therefore, the plant must enter a cold condition within 24 hours once more than 2 SRVs are determined to be inoperable. (See reload evaluation for the current cycle).

A manual actuation of each SRV is performed to demonstrate that the valves are mechanically functional and that no blockage exists in the valve discharge line. Valve opening is confirmed by monitoring the response of the turbine bypass valves and the SRV acoustic monitors. Adequate reactor steam dome pressure must be available to avoid damaging the valve. Adequate steam flow is required to ensure that reactor pressure can be maintained during the test. Testing is performed in the RUN mode to reduce the risk of a reactor scram in response to small pressure ~~flucuations~~ **fluctuations** which may occur while opening and reclosing the valves.

Low power physics testing and reactor operator training with inoperable components will be conducted only when the safety/relief valves are

3.7 (cont'd)

(2) During testing which adds heat to the suppression pool, the water temperature shall not exceed 10°F above the normal power operation limit specified in (1) above. In connection with such testing, the pool temperature must be reduced to below the normal power operation limit specified in (1) above within 24 hours.

(3) The reactor shall be scrammed from any operating condition if the pool temperature reaches 110°F. Power operation shall not be resumed until the pool temperature is reduced below the normal power operation limit specified in (1) above.

(4) During reactor isolation conditions, the reactor pressure vessel shall be depressurized to less than 200 psig at normal cooldown rates if the pool temperature reaches 120°F.

2. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F, and fuel is in the reactor vessel, except while performing low power physics tests at atmospheric pressure at power levels not to exceed 5 MWt.

4.7 (cont'd)

2. a. Perform required visual examination and leakage rate testing of the Primary Containment in accordance with the Primary Containment Leakage Rate Testing Program.

b. Demonstrate leakage rate through each MSIV is ≤ 11.5 scfh when tested at ≥ 25 psig. The testing frequency is in accordance with the Primary Containment Leakage Rate Testing Program.

c. Once per 24 months, demonstrate the leakage rate of 10AOV-68A,B for the Low Pressure Coolant Injection system and 14AOV-13A,B for the Core Spray system to be less than 11 scfm per valve when pneumatically tested at ≥ 45 psig at ambient temperature, or less than 10 gpm per valve if hydrostatically tested at ≥ ~~1000~~ **1,035** psig at ambient temperature.

3.7 BASES (cont'd)

Using the minimum or maximum torus water level (which are based on downcomer submergence levels where 13.88 feet above the bottom of the torus is 0.005 feet higher than the minimum submergence of 51.5 inches and 14.00 feet above the bottom of the torus is equivalent to the maximum submergence of 53 inches assumed in containment analyses) 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 inches results in a minimum torus water volume of approximately ~~105,600~~ **105,900** feet^3. The majority of the Bodega tests (9) were run with a submerged length of 4 feet 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 torus 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 limit for~~ complete condensation of the reactor coolant, although condensation would occur for temperatures above 170°F.

INSERT A

~~Using a 40°F rise (Section 5.2 FSAR) in the torus 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.~~

~~For an initial maximum torus 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 (NPSH) for the core spray LPCI and HPCI pumps.~~

Limiting suppression pool temperature to ~~130~~ **105°F** during RCIC, HPCI, or relief valve operation, when decay heat and stored energy are removed ~~form~~ **from** the primary system by discharging reactor steam directly to the torus assures adequate margin for a potential blowdown any time during RCIC, HPCI, or relief valve operation.

INSERT B

~~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 torus loadings.~~

Changes to Technical Specification Page 188

INSERT A

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 torus 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 B

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 SRV's 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.

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.

~~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.~~ **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 evaluations, References 19, 20, and 21, assumed: the primary containment leak rate was 1.5 volume percent per day; source term releases were in accordance with TID-14844 and Regulatory Guide 1.3, and were consistent with the Standard Review Plan; and the standby gas treatment system filter efficiency was 99% for halogens.** These doses are also based on the

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) Deleted

(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, ~~Progress Report for Period Ending December 31, 1966,~~ ORNL-4071."

(11) Section 5.2 of the FSAR.

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

(13) Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program", dated September 1995.

(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) 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors, Option B - Performance Based Requirements", Effective Date October 26, 1995

(17) Deleted

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(18) General Electric Report NEDC-32016P-1, "Power Uprate Safety Analysis for James A. FitzPatrick Nuclear Power Plant," April 1993 (proprietary), Including Errata and Addenda Sheet No. 1, dated January 1994.

(19) James A. FitzPatrick Calculation JAF-CALC-RAD-00023, Rev. 0, "Power Uprate Program - Technical Support Center Post-Accident Radiological Habitability Study," August 1996.

(20) James A. FitzPatrick Calculation JAF-CALC-RAD-00042, Rev. 0, "Control Room Radiological Habitability Under Power Uprate Conditions and CREVASS Reconfiguration," September 1995.

7.0 REFERENCES (continued)

(21) James A. FitzPatrick Calculation JAF-CALC-RAD-00048, Rev. 0, "Power Uprate Project - Radiological Impact at Onsite and Offsite Outdoor Receptors Following Design-Basis Accidents," May 1996

(22) General Electric Report GE-NE-187-45-1191, "Containment Systems Evaluation for the James A. FitzPatrick Nuclear Power Plant," November 1991 (proprietary).

Attachment IV to JPN-96-046

ERRATA AND ADDENDA SHEET NO. 1, DATED JANUARY 1994
FOR
GENERAL ELECTRIC REPORT NEDC-32016P-1

New York Power Authority

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

ERRATA AND ADDENDA SHEET NO. 1, JANUARY 1994

FOR

GENERAL ELECTRIC REPORT NEDC-32016P-1

The following corrections apply to Table 5-1 (page 5-8) of General Electric Report NEDC-32016P-1, "Power Uprate Safety Analysis for James A. FitzPatrick Nuclear Power Plant," dated April 1993.

For the parameter listed as "Vessel High Pressure Scram":

1) The value "1045 psig" should be "1059 psig"

2) The value "1080 psig" should be "1094 psig"

The values 1045 psig and 1080 psig are Technical Specification limits. The corrected values, 1059 psig and 1094 psig respectively, are the analytical limits which were intended to be reported in this table.

The text in this report (Section 5.1.2.1, page 5-3) associated with Table 5-1 is correct as written.