Entergy

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Robert G. Smith, P.E.
Site Vice President

February 18, 2011

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555-0001

SUBJECT: Entergy Nuclear Operations, Inc.
Pilgrim Nuclear Power Station
Docket No. 50-293
License No. DPR-35

Marked-up and Re-typed Technical Specification and Bases Pages for the Proposed License Amendment Related to Setpoint and Setpoint Tolerance Increases for Safety Relief Valves (SRV) and Spring Safety Valves (SSV) (TAC No. ME3543)

REFERENCE:

1. Entergy Letter No. 2.10.016, Proposed License Amendment to Technical Specifications: Revised Technical Specification for Setpoint and Setpoint Tolerances Increases for Safety Valves (SRV) and Spring Safety Valves (SSV), and Related Changes, dated March 15, 2010.

2. Entergy Letter No. 2.11.007, Entergy Response to NRC Request for Additional Information dated January 10, 2011, to support the review of Setpoint and Setpoint Tolerance Increases for Safety Relief Valves (SRV) and Spring Safety Valves (SSV) (TAC No. ME3543), dated January 31, 2011.

LETTER NUMBER: 2.11.013

Dear Sir or Madam,

This letter provides the final marked-up and Re-typed Technical Specification pages and Bases in support of the proposed License Amendment requested by Reference 1, based on Entergy's response to NRC Request for Additional Information submitted by Reference 2.

Attachment 1: Marked-up Technical Specification Pages and Bases

Attachment 2: Re-typed Technical Specification Pages and Bases

These Attachments supersede the prior marked-up and re-typed Technical Specification and Bases provided by Reference 1 and 2, respectively.

A001
NRR

These attachments support the proposed License Amendment for Setpoint and Setpoint Tolerance Increases for Safety Relief Valves (SRV) and Spring Safety Valves (SSV), and Related Changes and the No Significant Hazards Consideration determination submitted by Reference 1.

There are no commitments made in this submittal.

If you have any questions, please call Mr. Joseph Lynch, Pilgrim Licensing Manager at 508-830-8403.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 18th day of February, 2011

Sincerely,

For RGS

Robert Smith
Site Vice President

Attachment 1: Marked-up Technical Specification and Bases Pages (18 pages)

Attachment 2: Re-Typed Technical Specification and Bases Pages (14 Pages)

CC: W/O Attachments

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ATTACHMENT 1

To Entergy Letter No. 2.11.013

MARKED-UP TECHNCIAL SPECIFICATION AND BASES PAGES

(18 Pages)

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B. Technical Specifications

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

C. Records

ENO shall keep facility operating records in accordance with the requirements of the Technical Specifications.

D. Equalizer Valve Restriction - DELETED

E. Recirculation Loop Inoperable - DELETED

F. Fire Protection

ENO shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility and as approved in the SER dated December 21, 1978 as supplemented subject to the following provision:

ENO may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

G. Physical Protection

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contain Safeguards Information protected under 10 CFR 73.21, is entitled: "Pilgrim Nuclear Power Station Physical Security, Training and Qualification, and Safeguards Contingency Plan, Revision 0" submitted by letter dated October 13, 2004, as supplemented by letter dated May 15, 2006.

2.0 SAFETY LIMITS

2.1 Safety Limits

2.1.1 With the reactor steam dome pressure < 785 psig or core flow < 10% of rated core flow:

THERMAL POWER shall be ≤ 25% of RATED THERMAL POWER.

2.1.2 With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% of rated core flow:

MINIMUM CRITICAL POWER RATIO shall be ≥ 1.08 for two recirculation loop operation or ≥ 1.11 for single recirculation loop operation.

add space

2.1.3 Whenever the reactor is in the cold shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than 12 inches above the top of the normal active fuel zone.

2.1.4 Reactor steam dome pressure shall be ≤ ~~1325~~ psig at any time when irradiated fuel is present in the reactor vessel.

1340

2.2 Safety Limit Violation

With any Safety Limit not met within two hours the following actions shall be met:

2.2.1 Restore compliance with all Safety Limits, and

2.2.2 Insert all insertable control rods.

BASES:

See insert A

2.0 SAFETY LIMITS (Cont)

REACTOR STEAM DOME PRESSURE (2.1.4)

~~The Safety Limit for the reactor steam dome pressure has been selected such that it is at a pressure below which it can be shown that the integrity of the system is not endangered. The reactor pressure vessel is designed to Section III of the ASME Boiler and Pressure Vessel Code (1965 Edition, including the January 1966 Addendum), which permits a maximum pressure transient of 110%, 1375 psig, of design pressure 1250 psig. The Safety Limit of 1325 psig, as measured by the reactor steam dome pressure indicator, is equivalent to 1375 psig at the lowest elevation of the reactor coolant system. The reactor coolant system is designed to ASME Section III for the reactor recirculation piping, which permits a maximum pressure transient of 120% of design pressures of 1148 psig at 562°F for suction piping and 1241 psig at 562°F for discharge piping. The pressure Safety Limit is selected to be the lowest transient overpressure allowed by the applicable codes.~~

REFERENCES

1) "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A (through the latest approved amendment at the time the reload analyses are performed as specified in the CORE OPERATING LIMITS REPORT).

2) General Electric Thermal Analysis Basis (GETAB): Data, Correlation and Design Application, General Electric Co. BWR Systems Department, January 1977, NEDE-10958-PA and NEDO-10958-A.

3) "Methodology & Uncertainties for SLMCPR Evaluations," NEDC-32601-P-A (August 1999).

4) "Power Distribution Uncertainties for Safety Limit MCPR Evaluations," NEDC-32694-P-A (August 1999).

5) "GE 11 Compliance with Amendment 22 of GESTAR II," NEDE-31917P (April 1991).

6) "GE 14 Compliance with Amendment 22 of GESTAR II," NEDC-32868P (December 1998).

→ Add Insert B

INSERT A

The Safety Limit for the reactor steam dome pressure has been selected such that it is at a pressure below which it can be shown that the integrity of the reactor coolant system is not endangered. The reactor pressure limit of 1340 psig as measured in the vessel steam dome was derived from the design pressures of the reactor vessel. The peak pressures for the piping systems connected to the reactor vessel have been recalculated based on a reactor steam dome peak pressure of 1340 psig. These peak pressures are below the lowest of the transient pressures permitted by the applicable design code: ASME Boiler and Pressure Vessel (B&PV) Code (1965 Edition, including January 1966 Addendum) for the pressure vessel, USAS Piping Code Section B31.1 for the steam space piping, and ASME Section III for the reactor coolant system recirculation piping. The ASME B&PV Code permits pressure transients up to 10% over the vessel design pressure (110% x 1250 = 1375 psig). The USAS Piping Code and ASME Section III permit pressure transients and other occasional loads whose combined effect do not exceed stress levels based on the duration of the loads and the applicable service limit.

INSERT B

7) "Pilgrim Nuclear Power Station Safety Valve Setpoint Increase," GE Hitachi Nuclear Energy Report, NEDC-33532P, Rev. 2 (January 2011)

TABLE 3.2.F (Cont)

SURVEILLANCE INSTRUMENTATION

Minimum # of Channels	Operable Instrument Instrument #	Parameter	Type Indication and Range	Notes
2	TI-5021-2A TRU-5021-1A	Suppression Chamber Water Temperature	Indicator/ Multipoint Recorder 30-230°F (Bulk)	(1) (2) (3) (4)
	TI-5022-2B TRU-5022-1B	Suppression Chamber Water Temperature	Indicator/ Multipoint Recorder 30-230°F (Bulk)	(1) (2) (3) (4)
1	PID-5021	Drywell/Torus Diff. Pressure	Indicator -.25 - +3.0 psig	(1) (2) (3) (4)
1	PID-5067A PID-5067B	Drywell Pressure Torus Pressure	Indicator -.25 - +3.0 psig Indicator - 1.0 - +2.0 psig	(1) (2) (3) (4)
1/Valve	(a)Primary or (b) Backup	Safety/Relief Valve Position	(a) Acoustic monitor (b) Thermocouple	(5)
1/Valve	(a)Primary or (b) Backup	Safety Valve Position Indicator	(a) Acoustic monitor (b) Thermocouple	(5)
~~1/Valve~~	~~See Note (6)~~	~~Tail Pipe Temperature Indication~~	~~Thermocouple~~	~~(6)~~
2	LI-1001-604A LR-1001-604A LI-1001-604B LR-1001-604B	Torus Water Level (Wide Range) Torus Water Level (Wide Range)	Indicator/Multipoint Recorder 0-300" H_2O Indicator/Multipoint Recorder 0-300" H_2O	(1) (2) (3) (4) (1) (2) (3) (4)

NOTES FOR TABLE 3.2.F

(1) With less than the minimum number of instrument channels, restore the inoperable channel(s) within 30 days.

(2) With the instrument channel(s) providing no indication to the control room, restore the indication to the control room within seven days.

(3) If the requirements of notes (1) or (2) cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the Cold Shutdown Condition within 24 hours.

(4) These surveillance instruments are considered to be redundant to each other.

(5) At a minimum, the primary or back-up parameter indicators shall be operable for each valve when the valves are required to be operable. With both primary and backup instrument channels inoperable either return one (1) channel to operable status within 31 days or be in a shutdown mode within 24 hours.

The following instruments are associated with the safety/relief and safety valves:

Valve	Primary Acoustic Monitor	~~Secondary~~ Backup Tail Pipe Temperature Thermocouple
203-3A	ZT-203-3A	TE6285 ~~TE6271~~ *
203-3B	ZT-203-3B	TE6286 ~~TE6272~~ *
203-3C	ZT-203-3C	TE6287 ~~TE6273~~ *
203-3D	ZT-203-3D	TE6288 ~~TE6274~~ *
203-4A	ZT-203-4A	TE6274-B
203-4B	ZT-203-4B	TE6275-B

~~* See Note (6)~~

deleted

(6) At a minimum, for thermocouples providing SRV tail pipe temperature, one of the dual thermocouples will be operable for each SRV when the valves are required to be operable. If a thermocouple becomes inoperable, it shall be returned to an operable condition within 31 days or the reactor shall be placed in a shutdown mode within 24 hours.

(7) With less than the minimum number of operable instrument channels, restore the inoperable channels to operable status within 7 days or prepare and submit a special report to the Commission within 14 days of the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the channels to operable status.

PNPS
TABLE 3.2-G

INSTRUMENTATION THAT INITIATES RECIRCULATION PUMP TRIP
AND
ALTERNATE ROD INSERTION

Minimum Number of Operable or Tripped Instrument Channels Per Trip System (1)	Trip Function	Trip Level Setting
2	High Reactor Dome Pressure	~~1175 ± 5~~ psig ≤ 1210
2	Low-Low Reactor Water Level	≥ -46.3" indicated level

Actions (1) There shall be two (2) operable trip systems for each function.

(a) If the minimum number of operable or tripped instrument channels for one (1) trip system cannot be met, restore the trip system to operable status within 14 days or be in at least hot shutdown within 24 hours.

(b) If the minimum operability conditions (1.a) cannot be met for both (2) trip systems, be in at least hot shutdown within 24 hours.

BASES:

3.2 PROTECTIVE INSTRUMENTATION (Cont)

The recirculation pump trip/alternate rod insertion systems are consistent with the "Monticello RPT/ARI" design described in NEDO-25016 (Reference 1) as referenced by the NRC as an acceptable design (Reference 2) for RPT. Reference 1 provides both system descriptions and performance analyses. The pump trip is provided to minimize reactor pressure in the highly unlikely event of a plant transient coincident with the failure of all control rods to scram. The rapid flow reduction increases core voiding providing a negative reactivity feedback. High pressure sensors and low water level sensors initiate the trip. The recirculation pump trip is only required at high reactor power levels, where the safety/relief valves have insufficient capacity to relieve the steam which continues to be generated in this unlikely postulated event. Requiring the trip to be operable only when in the RUN mode is therefore conservative. The low water level trip function includes a time delay of nine (9) seconds ± one (1) second to avoid increasing the consequences of a postulated LOCA. This delay has an insignificant effect on ATWS consequences.

Alternate rod insertion utlizes the same initiation logic and functions as RPT and provides a diverse means of initiating a reactor scram. ARI uses sensors diverse from the reactor protection system to depressurize the scram pilot air header, which in turn causes all control rods to be inserted.

Additional analysis of the ARI/RPT setpoint for High Reactor Dome Pressure is identified in Reference 3

References

1. NEDO-25016, "Evaluation of Anticipated Transients Without Scram for the Monticello Nuclear Generating Plant," September 1976.

2. NUREG-0460, Volume 3, December 1978.

3. "Pilgrim Nuclear Power Station Safety Valve Setpoint Increase," GE-Hitachi Nuclear Energy Report NEDC-33532P, Rev. 2, January 2011

Drywell Temperature

The drywell temperature limitations of Specification 3.2.H.1 ensure that safety related equipment will not be subjected to excess temperature. Exposure to excessive temperatures may degrade equipment and can cause loss of its operability.

The temperature elements for monitoring drywell temperature specified in Table 3.2.H were chosen on the basis of their reliability, location, and their redundancy (dual - element RTD's). These temperature elements are the primary elements used for the PCILRT.

The "nominal instrument elevations" provided in Tables 3.2.H and 4.2.H assist personnel in locating the instruments for surveillance and maintenance purposes and define the approximate containment region to be monitored. The "nominal instrument elevations" are not intended to provide a precise instrument location.

LIMITING CONDITIONS FOR OPERATION

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

C. HPCI System

1. The HPCI system shall be operable whenever there is irradiated fuel in the reactor vessel, reactor pressure is greater than 150 psig., and reactor coolant temperature is greater than 365°F, except as specified in 3.5.C.2 below.

2. From and after the date that the HPCI system is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding 14 days unless such system is sooner made operable, providing that during such 14 days all active components of the ADS system, the RCIC system, the LPCI system and both core spray systems are operable.

3. If the requirements of 3.5.C cannot be met, an orderly shutdown of the reactor shall be initiated and the reactor shall be in the Cold Shutdown Condition within 24 hours.

SURVEILLANCE REQUIREMENTS

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

C. HPCI System

1. HPCI system testing shall be as follows:

a.	Simulated Automatic Actuation Test	Once/ Operating Cycle (See insert C)
b.	Pump Operability	When tested as specified in 3.13, verify that the HPCI pump delivers at least 4250 GPM for a system head corresponding to a reactor pressure of 1000 psig.
c.	Motor Operated Valve Operability	As Specified in 3.13 (See insert D)
d.	Flow Rate at 150 psig.	Once/ operating cycle, verify that the HPCI pump delivers at least 4250 GPM for a system head corresponding to a reactor pressure of 150 psig.

The HPCI pump shall deliver at least 4250 GPM for a system head corresponding to a reactor pressure of 1000 to 150 psig.

INSERT C

---------- Note ----------

Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform test.

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b. Pump Operability	When tested as specified in 3.13, verify with reactor pressure ≤ 1035 and ≥ 940 psig, the HPCI pump can develop a flow rate ≥ 4250 gpm against a system head corresponding to reactor pressure.

INSERT D

---------- Note ----------

Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform test.

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d. Flow rate at 150 psig	Once / operating cycle, verify with reactor pressure ≤ 150 psig, the HPCI pump can develop a flow rate ≥ 4250 gpm against a system head corresponding to reactor pressure.

LIMITING CONDITIONS FOR OPERATION

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

D. Reactor Core Isolation Cooling (RCIC) System

1. The RCIC system shall be operable whenever there is irradiated fuel in the reactor vessel, reactor pressure is greater than 150 psig, and reactor coolant temperature is greater than 365°F, except as specified in 3.5.D.2 below.

2. From and after the date that the RCIC system is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding 14 days unless such system is sooner made operable, providing that during such 14 days the HPCIS is operable.

3. If the requirements of 3.5.D cannot be met, an orderly shutdown of the reactor shall be initiated and the reactor shall be in the Cold Shutdown Condition within 24 hours.

SURVEILLANCE REQUIREMENTS

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

D. Reactor Core Isolation Cooling (RCIC) System

1. HPCI system testing shall be as follows:

a.	Simulated Automatic Actuation Test	Once/ Operating Cycle
b.	Pump Operability	When tested as specified in 3.13, verify that the RCIC pump delivers at least 400 GPM at a system head corresponding to a reactor pressure of 1000 psig.
c.	Motor Operated Valve Operability	As Specified in 3.13
d.	Flow Rate at 150 psig.	Once/ operating cycle verify that the RCIC pump delivers at least 400 GPM at a system head corresponding to a reactor pressure of 150 psig.

The RCIC pump shall deliver at least 400 GPM for a system head corresponding to a reactor pressure of 1000 to 150 psig.

See insert E

See Insert F

INSERT E

---------- Note ----------

Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform test.

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b. Pump Operability	When tested as specified in 3.13, verify with reactor pressure ≤ 1035 and ≥ 940 psig, the RCIC pump can develop a flow rate ≥ 400 gpm against a system head corresponding to reactor pressure.

INSERT F

---------- Note ----------

Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform test.

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d. Flow rate at 150 psig	Once / operating cycle, verify with reactor pressure ≤ 150 psig, the RCIC pump can develop a flow rate ≥ 400 gpm against a system head corresponding to reactor pressure.

B 3/4.5 CORE AND CONTAINMENT COOLING SYSTEMS

3/4.5.E. Automatic Depressurization (ADS) System
BASES

BACKGROUND

This specification ensures the operability of the ADS under all conditions for which the automatic or manual depressurization of the nuclear system is an essential response to station abnormalities.

The nuclear system pressure relief system provides automatic nuclear system depressurization for small breaks in the nuclear system so that the low pressure coolant injection (LPCI) and the core spray systems can operate to protect the fuel barrier.

Because the Automatic Depressurization System does not provide makeup to the reactor primary vessel, no credit is taken for the steam cooling of the core caused by the system actuation to provide further conservatism to the CSCS. Performance analysis of the Automatic Depressurization System is considered only with respect to its depressurizing effect in conjunction with LPCI or Core Spray. There are four valves provided and each has a capacity of ~~889,700~~ 921,235 lb/hr at a reactor pressure of ~~1115~~ 1155 psig.

APPLICABLE SAFETY ANALYSIS

The limiting conditions for operating the ADS are derived from the Station Nuclear Safety Operational Analysis (FSAR Appendix G) and a detailed functional analysis of the ADS (FSAR Section 6).

ACTIONS

The allowable out of service time for one ADS valve is determined as 14 days because of the redundancy and because of HPCI operability; therefore, redundant protection for the core with a small break in the nuclear system is still available.

SURVEILLANCES

The testing interval for the core and containment cooling systems is based on industry practice, quantitative reliability analysis, judgment and practicality. The core cooling systems have not been designed to be fully testable during operation. For example, complete ADS testing during power operation causes an undesirable loss-of-coolant inventory. When components are tested and found inoperable the impact on system operability is determined, and corrective action or Limiting Conditions of Operation are initiated. A simulated automatic actuation test once each cycle combined with code inservice testing of the pumps and valves is deemed to be adequate testing of these systems. The ADS test circuit permits continued surveillance on the operable relief valves to assure that they will be available if required.

The surveillance requirements provide adequate assurance that the core and containment cooling systems will be operable when required.

LIMITING CONDITIONS FOR OPERATION

3.6 PRIMARY SYSTEM BOUNDARY (Cont)

c. With no required leakage detection systems Operable, be in Hot Shutdown within the next 12 hours and in Cold Shutdown within the following 24 hours.

D. Safety and Relief Valves

1. During reactor power operating conditions and prior to reactor startup from a Cold Condition, or whenever reactor coolant pressure is greater than 104 psig and temperature greater than 340°F, both safety valves and the safety modes of all relief valves shall be operable. The nominal setpoint for the relief/safety valves shall be selected between 1095 and 1115 psig. All relief/safety valves shall be set at this nominal setpoint ± 11 psi. The safety valves shall be set at 1240 psig ± 13 psi.

2. If Specification 3.6.D.1 is not met, an orderly shutdown shall be initiated and the reactor coolant pressure shall be below 104 psig within 24 hours.

NOTE
Technical Specifications 3.6.D.3 - 3.6.D.5 apply to the two Stage Target Rock relief valves.

3. If the temperature of any safety relief discharge pipe exceeds 212°F during normal reactor power operation for a period of greater than 24 hours, an engineering evaluation shall be performed justifying continued operation for the corresponding temperature increases.

SURVEILLANCE REQUIREMENTS

4.6 PRIMARY SYSTEM BOUNDARY (Cont)

D. Safety and Relief Valves

1. Testing of safety and relief/safety valves shall be in accordance with 3.13.

2. At least one of the relief/safety valves shall be disassembled and inspected each refueling outage.

3. Whenever the safety relief valves are required to be operable, the discharge pipe temperature of each safety relief valve shall be logged daily.

4. Instrumentation shall be calibrated and checked as indicated in Table 4.2.F.

1. See Insert G

2. See Insert H

INSERT G

1. As specified in accordance with 3.13, verify the safety function lift setpoints of the safety and relief valves as follows:

Number of Safety and Relief Valves	Setpoint (psig)
2 Safety	1280 ± 38.4
4 Relief	1155 ± 34.6

Following testing, lift settings shall be within ± 1%.

INSERT H

---------- Note ----------
Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform test.
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2. Once / operating cycle, verify each relief valve opens when manually actuated.

LIMITING CONDITIONS FOR OPERATION

3.6 PRIMARY SYSTEM BOUNDARY (Cont)

D. Safety Relief Valves (Con't)

4. Any safety relief valve whose discharge pipe temperature exceeds 212°F for 24 hours or more shall be removed at the next cold shutdown of 72 hours or more, tested in the as-found condition, and recalibrated as necessary prior to reinstallation.

5. The limiting conditions of operation for the instrumentation that monitors tail pipe temperature are given in Table 3.2-F.

E. Jet Pumps

1. Whenever the reactor is in the Startup or Run Modes, all jet pumps shall be Operable. If it is determined that a jet pump is inoperable, the reactor shall be in Hot Shutdown within 12 hours.

SURVEILLANCE REQUIREMENTS

4.6 PRIMARY SYSTEM BOUNDARY (Cont)

E. Jet Pumps

NOTES

1. Not required to be performed until 4 hours after the associated recirculation loop is in operation.

2. Not required to be performed until 24 hours after >25% Rated Thermal Power.

Whenever there is recirculation flow with the reactor in the Startup or Run Modes, jet pump operability shall be checked daily by verifying at least one of the following criteria (1, 2, or 3) is satisfied for each operating recirculation loop:

1. Recirculation pump flow to speed ratio differs by ≤ 5% from established patterns, and jet pump loop flow to recirculation pump speed ratio differs by ≤ 5% from established patterns.

2. Each jet pump diffuser to lower plenum differential pressure differs by ≤ 20% from established patterns.

3. Each jet pump flow differs by ≤ 10% from established patterns.

BASES:

3/4.6 PRIMARY SYSTEM BOUNDARY (Cont)

C. Coolant Leakage (Cont)

The 2 gpm limit for unidentified coolant leakage rate increase within any 24 hour period is a limit specified by the NRC in Generic Letter 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping." This limit applies only during the RUN mode to accommodate the expected coolant leakage increase during pressurization.

The total leakage rate consists of all leakage, which flows to the drywell equipment drain sump (Identified leakage) and floor drain sump (Unidentified leakage).

In addition to the sump monitoring of coolant leakage, airborne radioactivity levels of the drywell atmosphere is monitored by the Reactor Pressure Boundary Leak Detection System. This system consists of two panels capable of monitoring the primary containment atmosphere for particulate and gaseous radioactivity as a result of coolant leaks.

D. Safety and Relief Valves

The valve sizing analysis considered four ~~11% capacity~~ relief/safety valves and two 8% capacity safety valves. These sizes and set pressures are established in accordance with the following three requirements of Section III of the ASME Code:

1. The lowest safety valve must be set to open at or below vessel design pressure and the highest safety valve be set at or below 105% of design pressure.

2. The valves must limit the reactor vessel pressure to no more than 110% of design pressure.

3. Protection systems directly related to the valve sizing transient must not be credited with action (i.e., an indirect scram must be assumed).

BASES:

3/4.6 PRIMARY SYSTEM BOUNDARY (Cont)

D. Safety and Relief Valves (Cont)

A main steam line isolation with flux scram has been selected to be used as the safety valve sizing transient since this transient results in the highest peak vessel pressure of any transient when analyzed with an indirect scram. The original FSAR analysis concluded that the peak pressure transient with indirect scram would be caused by a loss of condenser vacuum (turbine trip with failure of the bypass valves to open). However, later observations have shown that the long lengths of steam lines to the turbine buffer the faster stop valve closure isolation and thereby reduce the peak pressure caused by this transient to a value below that produced by a main steam line isolation with flux scram.

Item 3 above indicates that no credit be taken for the primary scram signal generated by closure of the main steam isolation valves. Two other scram initiation signals would be generated, one due to high neutron flux and one due to high reactor pressure. Thus item 3 will be satisfied by assuming a scram due to high neutron flux.

Relieving capacity of ~~44%~~ (4 relief/safety valves) in combination with ~~16%~~ (2 safety valves) results in a peak pressure during the transient conditions used in the safety valve sizing analysis which is well below the pressure safety limit.

The relief/safety valve settings satisfy the Code requirements that the lowest safety valve set point be at or below the vessel design pressure range to prevent unnecessary cycling caused by minor transients. The results of postulated transients where inherent relief/safety valve actuation is required are ~~given in Appendices R and Q of~~ the Final Safety Analysis Report.

Experience in safety valve operation shows that a testing of at least 50% of the safety valves per refueling outage is adequate to detect failures or deterioration. The tolerance value of ± ~~1%~~ is in accordance with Section III of the ASME Boiler and Pressure Vessel Code. An analysis has been performed which shows that with all safety valves set ~~1%~~ higher, the reactor coolant pressure safety limit of 1375 psig is not exceeded.

The relief/safety valves have two functions; i.e., power relief or self-actuated by high pressure. Power relief is a solenoid actuated function (Automatic Pressure Relief) in which external instrumentation signals of coincident high drywell pressure and low-low water level initiate the valves to open. This function is discussed in Specification 3.5~~D~~. In addition, the valves can be operated manually.

~~Pilgrim's experience with 2 stage safety/relief valves has demonstrated that minimum leakage exists when the tailpipe temperature is 215° Fahrenheit.~~

ATTACHMENT 2

To Entergy Letter No. 2.11.013

RE-TYPED TECHNCIAL SPECIFICATION AND BASES PAGES

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B. Technical Specifications

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

C. Records

ENO shall keep facility operating records in accordance with the requirements of the Technical Specifications.

D. Equalizer Valve Restriction - DELETED

E. Recirculation Loop Inoperable - DELETED

F. Fire Protection

ENO shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility and as approved in the SER dated December 21, 1978 as supplemented subject to the following provision:

ENO may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

G. Physical Protection

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contain Safeguards Information protected under 10 CFR 73.21, is entitled: "Pilgrim Nuclear Power Station Physical Security, Training and Qualification, and Safeguards Contingency Plan, Revision 0" submitted by letter dated October 13, 2004, as supplemented by letter dated May 15, 2006.

2.0 SAFETY LIMITS

2.1 Safety Limits

2.1.1 With the reactor steam dome pressure < 785 psig or core flow < 10% of rated core flow:

THERMAL POWER shall be ≤ 25% of RATED THERMAL POWER.

2.1.2 With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% of rated core flow:

MINIMUM CRITICAL POWER RATIO shall be ≥ 1.08 for two recirculation loop operation or ≥ 1.11 for single recirculation loop operation.

2.1.3 Whenever the reactor is in the cold shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than 12 inches above the top of the normal active fuel zone.

2.1.4 Reactor steam dome pressure shall be ≤ 1340 psig at any time when irradiated fuel is present in the reactor vessel.

2.2 Safety Limit Violation

With any Safety Limit not met within two hours the following actions shall be met:

2.2.1 Restore compliance with all Safety Limits, and

2.2.2 Insert all insertable control rods.

REACTOR STEAM DOME PRESSURE (2.1.4)

The Safety Limit for the reactor steam dome pressure has been selected such that it is at a pressure below which it can be shown that the integrity of the reactor coolant system is not endangered. The reactor pressure limit of 1340 psig as measured in the vessel steam dome was derived from the design pressure of the reactor vessel. The peak pressures for the piping systems connected to the reactor vessel have been recalculated based on a reactor steam dome peak pressure of 1340 psig. These peak pressures are below the lowest of the transient pressures permitted by the applicable design code: ASME Boiler and Pressure Vessel (B&PV) Code (1965 Edition, including the January 1966 Addendum) for the pressure vessel, USAS Piping Code B31.1 for the steam space piping and ASME Section III for the reactor coolant system recirculation piping. The ASME B&PV Code permits pressure transients up to 10% over the design pressure (110% x1250=1375 psig). The USAS Piping Code and ASME Section III permit pressure transients and other occasional loads whose combined effect do not exceed stress levels based on the duration of the loads and the applicable service limit.

REFERENCES

1) "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A (through the latest approved amendment at the time the reload analyses are performed as specified in the CORE OPERATING LIMITS REPORT).

2) General Electric Thermal Analysis Basis (GETAB): Data, Correlation and Design Application, General Electric Co. BWR Systems Department, January 1977, NEDE-10958-PA and NEDO-10958-A.

3) "Methodology & Uncertainties for SLMCPR Evaluations," NEDC-32601-P-A (August 1999).

4) "Power Distribution Uncertainties for Safety Limit MCPR Evaluations," NEDC-32694-P-A (August 1999).

5) "GE 11 Compliance with Amendment 22 of GESTAR II," NEDE-31917P (April 1991).

6) "GE 14 Compliance with Amendment 22 of GESTAR II," NEDC-32868P (December 1998).

7) "Pilgrim Nuclear Power Station Safety Valve Setpoint Increase," GE Hitachi Nuclear Energy Report, NEDC-33532P, Rev. 2 (January 2011)

TABLE 3.2.F (Cont)

SURVEILLANCE INSTRUMENTATION

Minimum # of Channels	Operable Instrument Instrument #	Parameter	Type Indication and Range	Notes
2	TI-5021-2A TRU-5021-1A TI-5022-2B TRU-5022-1B	Suppression Chamber Water Temperature Suppression Chamber Water Temperature	Indicator/ Multipoint Recorder 30-230°F (Bulk) Indicator/ Multipoint Recorder 30-230°F (Bulk)	(1) (2) (3) (4)) (1) (2) (3) (4)
1	PID-5021	Drywell/Torus Diff. Pressure	Indicator - .25 - +3.0 psig	(1) (2) (3) (4)
1	PID-5067A PID-5067B	Drywell Pressure Torus Pressure	Indicator -.25 - +3.0 psig Indicator -1.0 - +2.0 psig	(1) (2) (3) (4)
1/Valve	(a) Primary or (b) Backup	Safety/Relief Valve Position	a) Acoustic monitor b) Thermocouple	(5)
1/Valve	(a) Primary or (b) Backup	Safety Valve Position Indicator	a) Acoustic monitor b) Thermocouple	(5)
2	LI-1001-604A LR- 1001-604A LI-1001- 604B LR-1001- 604B	Torus Water Level (Wide Range) Torus Water Level (Wide Range)	Indicator /Multipoint Recorder 0 - 300"H_2O Indicator /Multipoint Recorder 0 - 300"H_2O	(1) (2) (3) (4) (1) (2) (3) (4)

NOTES FOR TABLE 3.2.F

(1) With less than the minimum number of instrument channels, restore the inoperable channel(s) within 30 days.

(2) With the instrument channel(s) providing no indication to the control room, restore the indication to the control room within seven days.

(3) If the requirements of notes (1) or (2) cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the Cold Shutdown Condition within 24 hours.

(4) These surveillance instruments are considered to be redundant to each other.

(5) At a minimum, the primary or backup parameter indicators shall be operable for each valve when the valves are required to be operable. With both primary and backup instrument channels inoperable either return one (1) channel to operable status within 31 days or be in a shutdown mode within 24 hours.

The following instruments are associated with the safety/relief and safety valves:

Valve	Primary Acoustic Monitor	Backup Tail Pipe Temperature Thermocouple
203-3A	ZT-203-3A	TE6285
203-3B	ZT-203-3B	TE6286
203-3C	ZT-203-3C	TE6287
203-3D	ZT-203-3D	TE6288
203-4A	ZT-203-4A	TE6274-B
203-4B	ZT-203-4B	TE6275-B

(6) Deleted.

(7) With less than the minimum number of operable instrument channels, restore the inoperable channels to operable status within 7 days or prepare and submit a special report to the Commission within 14 days of the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the channels to operable status.

TABLE 3.2-G

INSTRUMENTATION THAT INITIATES RECIRCULATION PUMP TRIP
AND
ALTERNATE ROD INSERTION

Minimum Number of Operable or Tripped Instrument Channels Per Trip System (1)	Trip Function	Trip Level Setting
2	High Reactor Dome Pressure	≤1210 psig
2	Low-Low Reactor Water Level	≥-46.3" indicated level

Actions (1) There shall be two (2) operable trip systems for each function.

(a) If the minimum number of operable or tripped instrument channels for one (1) trip system cannot be met, restore the trip system to operable status within 14 days or be in at least hot shutdown within 24 hours.

(b) If the minimum operability conditions (1.a) cannot be met for both (2) trip systems, be in at least hot shutdown within 24 hours.

3.2 PROTECTIVE INSTRUMENTATION (Cont)

The recirculation pump trip/alternate rod insertion systems are consistent with the "Monticello RPT/ARI" design described in NEDO-25016 (Reference 1) as referenced by the NRC as an acceptable design (Reference 2) for RPT. Reference 1 provides both system descriptions and performance analyses. The pump trip is provided to minimize reactor pressure in the highly unlikely event of a plant transient coincident with the failure of all control rods to scram. The rapid flow reduction increases core voiding providing a negative reactivity feedback. High pressure sensors and low water level sensors initiate the trip. The recirculation pump trip is only required at high reactor power levels, where the safety/relief valves have insufficient capacity to relieve the steam which continues to be generated in this unlikely postulated event. Requiring the trip to be operable only when in the RUN mode is therefore conservative. The low water level trip function includes a time delay of nine (9) seconds ± one (1) second to avoid increasing the consequences of a postulated LOCA. This delay has an insignificant effect on ATWS consequences. Additional analysis of the ARI/RPT Setpoint for High Reactor Dome Pressure is identified in Reference 3.

Alternate rod insertion utlizes the same initiation logic and functions as RPT and provides a diverse means of initiating a reactor scram. ARI uses sensors diverse from the reactor protection system to depressurize the scram pilot air header, which in turn causes all control rods to be inserted.

References

1. NEDO-25016, "Evaluation of Anticipated Transients Without Scram for the Monticello Nuclear Generating Plant," September 1976.

2. NUREG-0460, Volume 3, December 1978.

3. "Pilgrim Nuclear Power Station Safety Valve Setpoint Increase," GE Hitachi Nuclear Energy Report, NEDC-33532P, Rev. 2, January 2011.

Drywell Temperature

The drywell temperature limitations of Specification 3.2.H.1 ensure that safety related equipment will not be subjected to excess temperature. Exposure to excessive temperatures may degrade equipment and can cause loss of its operability.

The temperature elements for monitoring drywell temperature specified in Table 3.2.H were chosen on the basis of their reliability, location, and their redundancy (dual - element RTD's). These temperature elements are the primary elements used for the PCILRT.

The "nominal instrument elevations" provided in Tables 3.2.H and 4.2.H assist personnel in locating the instruments for surveillance and maintenance purposes and define the approximate containment region to be monitored. The "nominal instrument elevations" are not intended to provide a precise instrument location.

LIMITING CONDITIONS FOR OPERATION

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

C. HPCI System

1. The HPCI system shall be operable whenever there is irradiated fuel in the reactor vessel, reactor pressure is greater than 150 psig., and reactor coolant temperature is greater than 365°F, except as specified in 3.5.C.2 below.

2. From and after the date that the HPCI system is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding 14 days unless such system is sooner made operable, providing that during such 14 days all active components of the ADS system, the RCIC system, the LPCI system and both core spray systems are operable.

3. If the requirements of 3.5.C cannot be met, an orderly shutdown of the reactor shall be initiated and the reactor shall be in the Cold Shutdown Condition within 24 hours.

SURVEILLANCE REQUIREMENTS

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

C. HPCI System

1. HPCI system testing shall be as follows:

a. Simulated Automatic Actuation Test	Once/ Operating Cycle

---------- Note ----------
Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform test.
---

b. Pump Operability	When tested as specified in 3.13, verify with reactor pressure ≤1035 and ≥ 940 psig, the HPCI pump can develop a flow rate ≥ 4250 gpm against a system head corresponding to reactor pressure.
c. Motor Operated Valve Operability	As Specified in 3.13

---------- Note ----------
Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform test.
---

d. Flow Rate at 150 psig.	Once/ Operating Cycle, verify with reactor pressure ≤ 150 psig, the HPCI pump can develop a flow rate ≥ 4250 gpm against a system head corresponding to reactor pressure.

LIMITING CONDITIONS FOR OPERATION

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

D. Reactor Core Isolation Cooling (RCIC) System

1. The RCIC system shall be operable whenever there is irradiated fuel in the reactor vessel, reactor pressure is greater than 150 psig, and reactor coolant temperature is greater than 365°F, except as specified in 3.5.D.2 below.

2. From and after the date that the RCIC system is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding 14 days unless such system is sooner made operable, providing that during such 14 days the HPCIS is operable.

3. If the requirements of 3.5.D cannot be met, an orderly shutdown of the reactor shall be initiated and the reactor shall be in the Cold Shutdown Condition within 24 hours.

SURVEILLANCE REQUIREMENTS

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

D. Reactor Core Isolation Cooling (RCIC) System

1. RCIC system testing shall be as follows:

a. Simulated Automatic Actuation Test	Once/ Operating Cycle

---------- Note ----------
Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform test.
---

b. Pump Operability	When tested as specified in 3.13, verify with reactor pressure ≤ 1035 and ≥ 940 psig, the RCIC pump can develop a flow rate ≥ 400 gpm against a system head corresponding to reactor pressure.
c. Motor Operability Valve Operability	As Specified in 3.13

---------- Note ----------
Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform test.
---

d. Flow Rate at 150 psig.	Once/Operability Cycle verify with reactor pressure ≤ 150 psig, the RCIC pump can develop a flow rate ≥ 400 gpm against a system head corresponding to reactor pressure.

B 3/4.5 CORE AND CONTAINMENT COOLING SYSTEMS

3/4.5.E Automatic Depressurization (ADS) System
BASES

BACKGROUND

This specification ensures the operability of the ADS under all conditions for which the automatic or manual depressurization of the nuclear system is an essential response to station abnormalities.

The nuclear system pressure relief system provides automatic nuclear system depressurization for small breaks in the nuclear system so that the low pressure coolant injection (LPCI) and the core spray systems can operate to protect the fuel barrier.

Because the Automatic Depressurization System does not provide makeup to the reactor primary vessel, no credit is taken for the steam cooling of the core caused by the system actuation to provide further conservatism to the CSCS. Performance analysis of the Automatic Depressurization System is considered only with respect to its depressurizing effect in conjunction with LPCI or Core Spray. There are four valves provided and each has a capacity of 921,235 lb/hr at a reactor pressure of 1155 psig.

APPLICABLE SAFETY ANALYSIS

The limiting conditions for operating the ADS are derived from the Station Nuclear Safety Operational Analysis (FSAR Appendix G) and a detailed functional analysis of the ADS (FSAR Section 6).

ACTIONS

The allowable out of service time for one ADS valve is determined as 14 days because of the redundancy and because of HPCI operability; therefore, redundant protection for the core with a small break in the nuclear system is still available.

SURVEILLANCES

The testing interval for the core and containment cooling systems is based on industry practice, quantitative reliability analysis, judgment and practicality. The core cooling systems have not been designed to be fully testable during operation. For example, complete ADS testing during power operation causes an undesirable loss-of-coolant inventory. When components are tested and found inoperable the impact on system operability is determined, and corrective action or Limiting Conditions of Operation are initiated. A simulated automatic actuation test once each cycle combined with code inservice testing of the pumps and valves is deemed to be adequate testing of these systems. The ADS test circuit permits continued surveillance on the operable relief valves to assure that they will be available if required.

The surveillance requirements provide adequate assurance that the core and containment cooling systems will be operable when required.

LIMITING CONDITIONS FOR OPERATION

3.6 PRIMARY SYSTEM BOUNDARY (Cont)

c. With no required leakage detection systems Operable, be in Hot Shutdown within the next 12 hours and in Cold Shutdown within the following 24 hours.

D. Safety and Relief Valves

1. During reactor power operating conditions and prior to reactor startup from a Cold Condition, or whenever reactor coolant pressure is greater than 104 psig and temperature greater than 340°F, both safety valves and the safety modes of all relief valves shall be operable.

2. If Specification 3.6.D.1 is not met, an orderly shutdown shall be initiated and the reactor coolant pressure shall be below 104 psig within 24 hours.

SURVEILLANCE REQUIREMENTS

4.6 PRIMARY SYSTEM BOUNDARY (Cont)

D. Safety and Relief Valves

1. As specified in accordance with 3.13, verify the safety function lift setpoints of the safety and relief valves as follows:

No. of S/R Valves	Setpoint (psig)
2 Safety	1280 ± 38.4
4 Relief	1155 ± 34.6

Following testing, lift setting shall be within ± 1%.

---------- Note ----------
Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform test.
---

2. Once/ Operating Cycle, verify each relief valve opens when manually actuated.

LIMITING CONDITIONS FOR OPERATION

3.6 PRIMARY SYSTEM BOUNDARY (Cont)

E. Jet Pumps

1. Whenever the reactor is in the Startup or Run Modes, all jet pumps shall be Operable. If it is determined that a jet pump is inoperable, the reactor shall be in Hot Shutdown within 12 hours.

SURVEILLANCE REQUIREMENTS

4.6 PRIMARY SYSTEM BOUNDARY (Cont)

E. Jet Pumps

NOTES

1. Not required to be performed until 4 hours after the associated recirculation loop is in operation.
2. Not required to be performed until 24 hours after >25% Rated Thermal Power.

Whenever there is recirculation flow with the reactor in the Startup or Run Modes, jet pump operability shall be checked daily by verifying at least one of the following criteria (1, 2, or 3) is satisfied for each operating recirculation loop:

1. Recirculation pump flow to speed ratio differs by ≤ 5% from established patterns, and jet pump loop flow to recirculation pump speed ratio differs by ≤ 5% from established patterns.
2. Each jet pump diffuser to lower plenum differential pressure differs by ≤ 20% from established patterns.
3. Each jet pump flow differs by ≤ 10% from established patterns.

BASES:

3/4.6 PRIMARY SYSTEM BOUNDARY (Cont)

C. Coolant Leakage (Cont)

The 2 gpm limit for unidentified coolant leakage rate increase within any 24 hour period is a limit specified by the NRC in Generic Letter 88-01: "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping". This limit applies only during the RUN mode to accommodate the expected coolant leakage increase during pressurization.

The total leakage rate consists of all leakage, which flows to the drywell equipment drain sump (Identified leakage) and floor drain sump (Unidentified leakage).

In addition to the sump monitoring of coolant leakage, airborne radioactivity levels of the drywell atmosphere is monitored by the Reactor Pressure Boundary Leak Detection System. This system consists of two panels capable of monitoring the primary containment atmosphere for particulate and gaseous radioactivity as a result of coolant leaks

D. Safety and Relief Valves

The valve sizing analysis considered four relief/safety valves and two safety valves. The set pressures are established in accordance with the following three requirements of Section III of the ASME Code:

1. The lowest safety valve must be set to open at or below vessel design pressure and the highest safety valve be set at or below 105% of design pressure.

2. The valves must limit the reactor vessel pressure to no more than 110% of design pressure.

3. Protection systems directly related to the valve sizing transient must not be credited with action (i.e., an indirect scram must be assumed).

BASES:

3/4.6 PRIMARY SYSTEM BOUNDARY (Cont)

D. Safety and Relief Valves (Cont)

A main steam line isolation with flux scram has been selected to be used as the safety valve sizing transient since this transient results in the highest peak vessel pressure of any transient when analyzed with an indirect scram. The original FSAR analysis concluded that the peak pressure transient with indirect scram would be caused by a loss of condenser vacuum (turbine trip with failure of the bypass valves to open). However, later observations have shown that the long lengths of steam lines to the turbine buffer the faster stop valve closure isolation and thereby reduce the peak pressure caused by this transient to a value below that produced by a main steam line isolation with flux scram.

Item 3 above indicates that no credit be taken for the primary scram signal generated by closure of the main steam isolation valves. Two other scram initiation signals would be generated, one due to high neutron flux and one due to high reactor pressure. Thus item 3 will be satisfied by assuming a scram due to high neutron flux.

Relieving capacity of the 4 relief/safety valves in combination with the 2 safety valves results in a peak pressure during the transient conditions used in the safety valve sizing analysis which is well below the pressure safety limit.

The relief/safety valve settings satisfy the Code requirements that the lowest safety valve set point be at or below the vessel design pressure range to prevent unnecessary cycling caused by minor transients. The results of postulated transients where inherent relief/safety valve actuation is required are identified or referenced in the Updated Final Safety Analysis Report.

Experience in safety valve operation shows that a testing of at least 50% of the safety valves per refueling outage is adequate to detect failures or deterioration. The tolerance value of ± 3% is in accordance with Section III of the ASME Boiler and Pressure Vessel Code. An analysis has been performed which shows that with all safety valves set 3% higher, the reactor coolant system pressure safety limit of 1375 psig is not exceeded.

The relief/safety valves have two functions; i.e., power relief or self-actuated by high pressure. Power relief is a solenoid actuated function (Automatic Pressure Relief) in which external instrumentation signals of coincident high drywell pressure and low-low water level initiate the valves to open. This function is discussed in Specification 3.5.E. In addition, the valves can be operated manually.