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Perry Nuclear Power Plant
Docket No. 50-440
Response to Request for Additional Information Related to a License Amendment Requesting a Power Uprate (TAC No. MA6459)

Ladies and Gentlemen:

On September 9, 1999, the Perry Nuclear Power Plant (PNPP) staff submitted a license amendment request (PY-CEI/NRR-2420L) to the NRC requesting an increase in the present authorized rated thermal power level by 5% for the PNPP.

The PNPP staff received a Request for Additional Information (RAI) from the NRC dated January 27, 2000 regarding this license amendment request. The RAI was forwarded from the NRC Staff's Electrical and Instrumentation and Controls Branch, and the Reactor Systems Branch. The additional information requested is contained in Attachments 1 and 2, respectively.

In addition to the RAI responses, clarification is provided as requested within a February 8, 2000 conference call with the NRC staff on the resulting radiological dose considerations for the power uprate amendment request. This clarification is contained in Attachment 3. Also, Attachment 4 provides the applicable Technical Specification page annotated to reflect the revised setpoint as discussed in the response to the RAI forwarded by the NRC Staff's Electrical and Instrumentation and Controls Branch (#7).

If you have questions or require additional information, please contact Mr. Gregory A. Dunn, Manager - Regulatory Affairs, at (440) 280-5305.

Very truly yours,

John K Wood

Attachments

cc: NRC Project Manager
NRC Resident Inspector
NRC Region III
State of Ohio

A001

Perry Nuclear Power Plant Responses to an NRC Request For Additional Information (RAI) Forwarded by Electrical Section of the Electrical and Instrumentation and Controls Systems Branch

The Perry Plant staff received a Request for Additional Information (RAI) from the NRC dated January 27, 2000. The RAI deals with questions associated with the Perry Plant license amendment request regarding a proposed increase of the present authorized rated thermal power level (power uprate) for the Perry Plant. The following are responses to requests from the NRC's Electrical and Instrumentation and Controls Branch.

NRC QUESTION

1. In Section 6.1 of the Perry Plant power uprate submittal, it is noted that an offsite power grid stability uprate review determined the adequacy of the electrical equipment and grid stability. Please provide a concise description of what this grid stability uprate review consisted of and include in this description the major assumptions for this review and the resulting primary review findings and conclusions. In addition, please explain in detail what changes have been made to the relay protection systems for the 345 kV switchyard equipment and how those changes may affect the probability of losing electric power to the unit.

RESPONSE

The grid stability uprate review was performed to assess the impact of a proposed uprate of the Perry Plant on the transmission grid. The study examined system stability and reliability of off-site power. A variety of probable and severe scenarios, reflecting requirements contained in the National Electric Reliability Council (NERC) Planning Standards, Table I, were analyzed. These included single and double transmission line outages, single and double generating unit outages, and combination transmission line and generating unit outages. Both power flow and dynamic stability analyses were performed.

The power flow analysis considered thermal loading of transmission line and transformer branches and bus voltage violations under normal and contingency operating conditions. These were assessed relative to a benchmark model without the uprate. With the 5% uprate, no additional branch loading or bus voltage violations were observed, and no violations were intensified by the uprate. Stability analysis evaluated both first swing stability and system damping under contingencies. Responses to all the contingencies were stable and damped for both the benchmark and uprate models.

The model used in this analysis was designed to represent year 2000 summer conditions on the FirstEnergy system. It was based on a load flow model and dynamics data set prepared by the East Central Area Reliability Coordination Agreement (ECAR) Dynamic Analysis Working Group (DAWG) to represent 1999 Summer conditions.

This was the latest workable dynamics model set available at the time analysis was begun. The DAWG model was modified to increase loading for projected year 2000 conditions and provide more detailed representation of the FirstEnergy system - particularly in the vicinity of the Perry Plant.

The existing protective relay settings at the Perry Plant are based on full generator output of 1446 Mega Volt Amperes (MVA). Since the 5% uprate does not exceed 1446 MVA, no relay setting changes are required. Being that there were no changes in the relay settings, there is no change in the probability of losing electric power to the unit.

NRC QUESTION

2. Information provided in Section 6.1.1 of the subject submittal notes that the iso-phase bus ratings, the main power transformer ratings, and other associated switchyard component ratings (i.e., the unit and system auxiliary power transformer ratings and the generator current ratings) are adequate for the uprate operating conditions. Please provide the numerical rating values for each of these items and the expected numerical values for these items during operation at power uprated operating conditions. In addition, please explain the technical basis for the increase in the main transformers rating from 1394.4 MVA to 1580 MVA as described by Table 6-1.

RESPONSE

The following table provides the expected numerical values at uprated conditions compared to the existing ratings for the components requested.

Component	**Rating Units**	**Existing Value**	**Expected Value**
Generator	MVA	1446	1442 @ 0.91 PF
Isolated Phase Bus Duct	MVA	1524	1392
Main Transformers	MVA	1580	1392
Auxiliary Transformer	MVA	71	50
Switchyard (limiting)	MVA	1523	1392

The loadability of all of the switchyard components was reviewed. The lowest component loadability is for the 3500 kcmil aluminum wire drops. These wire drops are rated at 1523 MVA.

The uprated Perry Plant Main Transformer ratings identified were calculated using the Electric Power Research Institute (EPRI) transformer loadability program, "PTLoad – A Numerical Model for Power Transformer Load Planning." This program calculates a transformer's thermal response to hourly load and temperature changes over a 24 hour period based on the transformer's electrical and thermal characteristics.

The program calculates the maximum transformer MVA capability such that the specified maximum operating constraints of winding hot spot (110°C) and top oil temperature (110°C) are not exceeded. PTLoad facilitates the computation of ratings and thermal profiles for mineral-oil-immersed transformers based upon the calculation methods described in Institute of Electrical and Electronic Engineers (IEEE) C57.91-1995, "Guide for Loading Mineral-Oil-Immersed Transformers." The Perry Plant Main Transformer thermal characteristics were modeled using these basic principles and formulas based on test report (heat run) data supplied by the manufacturer and actual operating voltages. Historical ambient temperature data for FirstEnergy's service territory was used to model the environmental conditions.

The manufacturer's individual transformer nameplate rating was the basis for the original 1394.4 MVA total rating [3 @ 464.8 MVA Forced Oil and Air (FOA)/FOA 65°C]. The PTLoad calculations resulted in a total Main Transformer capability of 1580 MVA that utilizes the full thermal capabilities of the transformers under normal operating conditions while observing the hot spot and top oil temperature limits.

NRC QUESTION

3. Provide a discussion that addresses the impact of the power uprates on the load, voltage, and short circuit current values for all levels of the station auxiliary electrical distribution system (including ac and dc).

RESPONSE

The expected switchyard, generator, and battery voltage conditions are unchanged. The AC and DC electrical distribution configuration and characteristics are unchanged. Conservative load demand assumptions are used as the basis for the loading in the existing AC and DC design basis calculations. The only identifiable change in electrical load demand due to power uprate is associated with the Hotwell, Condensate Booster, and Feedwater Booster Pumps. These pumps experience increased flow due to uprated conditions. The increased pump demand was compared to the design basis motor demand assumed in the electrical system calculations. Since flows are only slightly increased, the motor demand for each of these loads remains bounded by the existing design basis calculations. All of the AC and DC system loads remain conservatively reflected in the existing design basis calculations. Since no electrical distribution or motor changes are planned and the load demand assumptions are conservative, no calculation changes are required, and the voltage and short circuit studies are unaffected.

NRC QUESTION

4. The subject submittal contains a discussion addressing how the proposed power uprate impacts the existing analysis performed for station blackout in Section 9.3.2. Please provide the numerical estimate for the increase in decay heat and associated temperature rise in the plant areas relevant to coping with station blackout conditions and discuss the potential impact of additional safety relief valve actuations due to the increased decay heat. Discuss and verify that the results of suppression pool temperature transient analyses show that emergency core cooling (ECCS) equipment will not be adversely impacted given a maximum allowable cooldown rate during the reactor pressure vessel depressurization. In general, quantify the changes including uncertainty bounds to the assumptions for the existing station blackout analysis under the power uprate conditions, particularly as they relate to issues such as heat-up analysis, equipment operability, and battery capacity.

RESPONSE

Numerical estimate for the increase in decay heat

Following the station blackout (SBO), decay heat increases roughly consistent with the degree of the uprate (i.e., approximately 5%). The SBO evaluation used a nominal decay heat for the uprated power level without additional uncertainty. Decay heat values at various times after shutdown were used.

Associated temperature rise in the plant areas relevant in coping with station blackout conditions

Only the suppression pool temperature was evaluated for the uprated SBO response. The results confirm that the suppression pool temperature remains below the 185 °F temperature limit. The temperature response in other plant areas, such as the battery area and High Pressure Core Spray (HPCS) room, is not expected to change due to the SBO event.

The potential impact of additional Safety Relief Valve actuations due to the increased decay heat

The increased decay heat will result in a slightly larger number of relief valve cycles prior to depressurization and an increase in the suppression pool temperature response. Since the pneumatic supply is sufficient during the SBO event and the low-low set logic is active, the number of SRV cycles is much lower than the design basis and there is no impact of the power uprate on SRV actuations.

Verify that ECCS will not be adversely impacted given a maximum allowable cooldown rate during the RPV depressurization

The only Emergency Core Cooling System (ECCS) equipment used during the SBO is the HPCS diesel and pump. HPCS operation is not impacted by the depressurization and the system is designed for operation at low reactor pressures. The HPCS pump Net Positive Suction Head (NPSH) can accommodate the increased suppression pool temperature.

Quantify the changes including uncertainty bounds to the assumptions for existing station blackout analysis under the power uprate conditions, particularly as they relate to issues such as heat-up analysis, equipment operability, and battery capacity

The SBO evaluation is a realistically-based evaluation using nominal values in accordance with Nuclear Management and Resources Council (NUMARC)-8700, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors" and Regulatory Guide 1.155, "Station Blackout." Consideration of uncertainty bounds is not appropriate for these analyses.

NRC QUESTION

5. In Section 10.3.1.1 of the subject submittal, it is stated that the current accident and normal plant conditions for temperature, pressure, and humidity inside the primary containment are "effectively unchanged" for the power uprate conditions. Please provide a detailed discussion to clearly explain how the current accident and normal temperature, pressure, and humidity profiles for inside the primary containment do change for the power uprate conditions and why these changes have no impact on the environment qualification of electrical equipment. In addition, please provide a similar discussion for the temperature, pressure, and humidity profiles for high energy line break areas outside of the primary containment.

RESPONSE

Inside Containment

Normal service temperatures are expected to increase little, if any, and there is no change in the normal operating pressure as a result of power uprate. Small changes in temperature do not affect qualification of electrical equipment but may affect the qualified life. The qualified life is addressed through the temperature monitoring program implemented at the Perry Plant. The two factors that affect relative humidity under normal operating conditions are temperature and leakage. Since the normal operating temperature in the drywell is expected to increase little with power uprate, and leakage into the drywell is not affected, it is concluded that drywell humidity remains unaffected.

Following an accident, relative humidity increases to 100% for the pre-uprate condition. Since this is the maximum value for relative humidity, there is no change for power uprate. There were changes to three of the uprate pressure response curves and one of the uprate temperature response curves, all inside containment that required evaluation.

Power uprate results in a peak containment pressure for the short-term Recirculation Suction Line Break (RSLB) and the short-term Main Steam Line Break (MSLB) of 22.84 psig and 23.45 psig, respectively. Each of these values exceeds the pre-uprate peak pressure of 21.8 psig, which is the criterion used to qualify equipment presently installed in the drywell. Examination of the System Component Evaluation Worksheets confirms that all components presently qualified for use inside the drywell are qualified to a peak pressure that exceeds 23.45 psig, plus the 10% margin suggested by IEEE 323-1974, "Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations" with one exception.

The hydrogen igniters were tested to a long-term pressure of 24.3 psig, which provides a margin of only 3.6% over the 23.45 psig value required for power uprate.

However, the peak pressure during this test was 32.6 psig, which envelopes the calculated peak pressure of 23.45 psig. Therefore, all components are qualified for containment pressure for the post-uprate conditions.

The post-uprate peak drywell temperature of 329°F (for an accident inside containment) did not increase above the pre-uprate qualification temperature of 330°F. Equipment qualification is based on the pre-uprate Small Line Break curve, which required a drywell peak temperature of 330°F for three hours, and 310°F for an additional three hours. The pre-uprate drywell temperature response curve is slightly more conservative compared to the post-uprate response curve, during the same time period. However, all components were tested to the more stringent Small Line Break curve, which bounds the post uprate drywell temperature response. Therefore, all components are qualified for the post-uprate peak drywell temperature.

The increase in peak drywell pressure and changes to the shape of the uprate pressure and temperature response curves discussed above were reviewed for potential impact on equipment qualification. It was concluded that these changes from the pre-uprate curves are minor and do not impact the qualification of equipment inside the drywell.

In summary, for power uprate, the calculated short term peak pressure used for qualification of equipment inside the drywell is greater than for pre-uprate calculations and there are minor changes to the shape of the pre-uprate temperature and pressure profiles. However, based on the evaluations performed, there is no impact on equipment qualification so these parameters are "effectively unchanged."

<u>HELB Areas Outside Containment</u>
The impact of uprated thermal power on the High Energy Line Break (HELB) transient outside containment was assessed using the GOTHIC computer code. The only HELB outside containment affected by power uprate is the rupture of a Reactor Water Cleanup (RWCU) line. The subsequent analysis demonstrates that the calculated temperature and pressure would not vary significantly (i.e., less than 0.1 psi and less than 1°F) with the uprate power blowdown. Therefore, the previous analyses remains valid for power uprate. Given that the previous analysis remains valid, there is no basis to change the qualification requirements of the equipment. Therefore, there is no impact on the environmental qualification of electrical equipment due to power uprate.

<u>NRC QUESTION</u>

6. In Sections 10.3.1.1 and 10.3.2 of the subject submittal, it is noted that the environmental qualification radiation levels under accident conditions are conservatively evaluated to increase 5% to 12% inside and outside the primary containment. It is also noted that the reevaluation of the environmental qualification conditions under the uprated power conditions identified some electrical equipment located inside the primary containment and mechanical equipment with non-metallic components which are affected by the higher accident radiation level.

Please identify this equipment and discuss how this equipment will be requalified for the new radiation values. Also provide the current, the revised, and bounding radiation level values and provide numerical values for specific equipment exposure under these new radiation conditions.

RESPONSE

It is important to note that for power uprate, under normal plant conditions, the radiation levels were evaluated to increase 5% for gamma and beta and 12% for neutron dose. For accident conditions, the evaluated increase in dose is 5% for gamma and beta.

After further detailed review, it was concluded that none of the equipment, which was previously identified as potentially affected, requires requalification. There were five (5) Auditable File Packages (AFPs) pursuant to 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants", where the equipment appeared to not have the required 10% accident margin (radiation) as a result of power uprate. However, after detailed review of these AFPs, it was confirmed that at least a 10% accident margin exists for the affected equipment. The first two AFPs identified involve the Power Range Detectors (PRD) and the Intermediate Range Detectors (IRD), respectively. The remaining AFPs identified involve the Residual Heat Removal (RHR) system pump motors, the Reactor Core Isolation Cooling (RCIC) Turbine Assembly, and the Fuel Handling Building (FHB) Ventilation System Exhaust Filter Plenums.

1) The PRDs, located in the reactor vessel, are only required to be operable 12 hours post accident for a small pipe break and 20 minutes for an Anticipated Transient Without Scram (ATWS) per the applicable environmental qualification report and the System Component Evaluation Worksheets (SCEW). Additionally, pursuant to controlled Perry Plant instructions, Perry Plant Reactor Engineering personnel calculate PRD "life" along with a life expectancy every six weeks. Therefore, the equipment qualification requirements of the PRDs are not impacted by the power uprate.

2) Pursuant to the SCEW and the applicable environmental qualification report, the IRDs, like the PRDs, are located in the reactor vessel and are only required for 12 hours following a small line break. The IRDs are only in use during plant start-up to approximately 12% power.

   Pursuant to the IRD qualification report, the designed life of the IRD is 1×10^{19} nv (neutron) and the plant operating neutron flux range is from 1×10^{8} to 1.5×10^{13} nv. The storage neutron flux is 5×10^{8} nv maximum. Additionally, it is identified in the IRD qualification report that the detectors are qualified for at least 15 years or until fissile material burn-up. Based on a 12% increase in neutron flux via the power uprate, the new values would be 1.12×10^{8} to 1.68×10^{13} nv for operation and 5.6×10^{8} nv for storage. Therefore, the maximum exposure would be $1.68 \times 10^{13} + 5.6 \times 10^{8} = 1.680056 \times 10^{13}$ nv. This is well below the design life of 1×10^{19} nv. Therefore, the equipment qualification requirements of the IRDs are not impacted by the power uprate.

3) The RHR pump motors are located in the Auxiliary Building, Equipment Qualification (EQ) Zone AB-4. For these motors, per the applicable environmental qualification report, the weak-link component with respect to radiation is silicone insulated leads. Therefore, the qualification requirements for the silicone insulated leads are bound by the other motor components. All other components are qualified to 2×10^8 Rads gamma (γ). As identified on the SCEW sheet for the motors and the applicable environmental qualification report, the motors are required for 100 days post LOCA. As such, the accident dose is 3.0×10^7 rads γ. Using the 100 day value of 3×10^7 rads γ as the accident dose, then the uprate to 105% would be 3.15×10^7 rads γ. This will result an acceptable margin pursuant to IEEE 323-1974 requirements for accident dose. Therefore, the equipment qualification requirements of the RHR pump motors are not impacted by the power uprate.

4) The RCIC Turbine is located in the Auxiliary Building, Equipment Qualification (EQ) Zone AB-4. As identified on the applicable SCEW sheet and the applicable environmental qualification report, the Total Integrated Dose (TID) for the RCIC Turbine Assembly is 1×10^4 rads γ. This value is below the required accident dose considering a "control rod drop" and ATWS event. The applicable environmental qualification report contains a detailed analysis that documents upgrading the RCIC turbine assembly to reflect the TID used for the HPCI turbine assembly. Although the Perry Plant does not have a HPCI Turbine, this is an equivalent component for comparison analysis per 10 CFR 50.49 and IEEE 323-1974 and requirements. Based on this information, when the TID for 40 years plus 6 hour accident are applied (1.1×10^6 rads), there remains an accident margin far in excess of IEEE 323-1974 requirements. Therefore, the equipment qualification requirements of the RCIC Turbine Assembly are not impacted by the power uprate.

5) The Fuel Handling Building (FHB) Ventilation system Exhaust Filter Plenums are located in the EQ Zones FB-7 and FB-8. It is identified in the SCEW that the limiting zone for an accident is Zone FB-8 with a dose of 1.52×10^6 rads γ. The Design Basis Accident (DBA) that requires activation of the Fuel Handling Building Ventilation system is a Fuel Handling Accident (FHA) of recently irradiated fuel. Perry Plant calculations identify the incremental doses and TID for a FHA, as listed on the applicable environmental zone drawing, to have a 10% margin above the actual values. Therefore, the equipment qualification requirements of the FHB Ventilation system Exhaust Filter Plenums are not impacted by the power uprate.

NRC QUESTION

7. The difference between the allowable value and the analytical limit for the Main Steamline High Flow Isolation (MSHLI) for the uprated power conditions represents a significant improvement in the setpoint determination given the known uncertainties and allowances specified in NEDC-31336, "General Electric Instrument Setpoint Methodology" dated October 1996. For example, NEDC-31336 specifies 1% allowance each for process measurement accuracy [BWR/6] and loop accuracy parameters and 2% allowance each for loop calibration and primary element accuracy parameters.

Please provide the calculation of the MSHLI instrument analytical limit and allowable value for the uprated power conditions with the current and revised steam flow, pressure and enthalpy conditions.

RESPONSE

The Main Steam Line Hi Flow Isolation setpoint for the power uprate condition has been recalculated. The calculation is based on an Analytical Limit (AL) of 140% of uprate flow and uses the Process Measurement Accuracy (PMA) allowance of 1% flow and Process Element Accuracy (PEA) allowance of 2% flow given in NEDC-31336 at the uprated AL flow condition. The resulting psid calculations of the AL and Allowable Value (AV) are 267.5 psid and 256.5 psid, respectively. Attachment 4 of this RAI response provides the revised Technical Specification change resulting from the revised AL for the Main Steam Line Hi Flow Isolation setpoint. Also, a copy of these calculations for the power uprate condition is included (see pages 10 through 17 of this RAI response).

MAIN STEAM LINE FLOW ELEMENT ANALYTICAL LIMIT (AL) CALCULATION

Inputs:

Input Description:		**Input Value**	**Reference(s)**
Mass Flow Rate, m^o	Lbm/hr	16.3e6 (I)	Ref. 4
Flow Pressure, P_o	psia	1040	Ref. 4
Flow Density, ρ	Lbm/ft^3	2.324 (II)	See Note II
Pipe Diameter, D	IN	23.358	Ref. 3
Throat Diameter, d_c	IN	11.88	Ref. 2
Area thermal-expansion factor, F_a	Ratio	1.0095	Ref. 1
Coefficient of discharge, C	Ratio	.995	Ref. 1

Notes:

I- Mass flow rate of 16.3 Mlbm/hr relates to 100% power uprate 3758 MWt and is for four steam lines. Mass flow rate per steam line is 4.075 Mlbm/hr. Analytical limit is calculated at 140% flow rate which is 1.4*4.075=5.705 Mlbm/hr.

II- Density is calculated at temperature of 552 ^oF and pressure of 1040 psia from ref. 3 and 4. Density related to saturated steam for 1040 psia is 2.343 lbm/ft^3 (at saturated temperature of 550 ^oF). However, the temperature for power uprate was assumed the same as pre-power uprate value because of no change in pressure.

Summary Results:

Analytical Limit (AL) is calculated for 100% power (3758 MWt) at 140 % flow rate. AL is 267.5 psid. Calculation for AL Performed in Accordance with Reference 1 methods and equations.

References:

1. ASME Research Committee, Fluid Meters, 6th Edition, 1971.

2. GE Drawing 105X5082P005, Rev. 7 " Flow Element", Upstream Casting Drawing.

3. Perry Main Steam Stress Report GE Document Number 23A6989 Rev 0, November 1998

4. Nominal Heat Balance GE-NE-A22-00084-01-01Rev. 0 Project Task Report for Perry Nuclear Power Plant Power Uprate Evaluation, December 1998.

1. **Function**: Main Steam Line Isolation – High MSL Flow

Setpoint Characteristics:	**Definition**	**Reference(s)**
Event Protection	Break in Main Steam Line	Ref. 5
Function After Earthquake	☒ Required ☐ Not Required	Ref. 2
Setpoint Direction	☒ Increasing ☐ Decreasing	Ref. 2
Single or Multiple Channel	☐ Single ☒ Multiple	Ref. 2
LER Calculation Basis if Multiple Channel	Standard (Conservative) LER Calculation☒ , or Configuration Specific LER Calculation ☐	Ref. 3
Trip Logic for Configuration Specific LER Calculation	N/A	

Current Function Limits:	**Value/Equation**	**Reference(s)**
Analytical Limit	193.6 PSID	Ref. 1
Tech Spec Allowable Value	189.3 PSID	Ref. 1
Setpoint	183 PSID	Ref. 1
Operational Limit	N/A	Ref. 1

Plant data:	**Value**	**Sigma if not 2**	**Reference(s)**
Primary Element • Accuracy (PEA) • Drift (sensitivity loss)	7.697 PSID (Random) 0.5 PSID (Bias)		Comment 1
Process Measurement Accuracy (PMA)	3.835 PSID (Bias)		Comment 2

Devices in Setpoint Function Instrument Loop:

- Differential Pressure Transmitter
- Trip Bi-stable Unit

2. Components:

2.1 Component: Differential Pressure Transmitter

Component Information:	Value/Equation	Reference(s)
Plant Instrument ID No.	1E31N086A,B,C,D 1E31N087A,B,C,D 1E31N088A,B,C,D 1E31N089A,B,C,D	Ref. 1
Instrument vendor	Rosemount	Ref. 2
Model ID No. (including Range Code)	1153DB7	Ref. 2
Plant Location(s)	Containment CO/08-620	Ref. 1
Process Element	Venturi Type Flow Element	Ref. 1

Inputs:

Vendor specifications:	Value / Equation	Sigma if not 2	Reference(s)
Top of Scale	300 PSID		Comment 3
Bottom of Scale	0 PSID		Comment 3
Upper Range Limit	300 PSID		Ref. 1
Accuracy	.25% OF SPAN	3	Ref. 1
Temperature Effect	1.754 PSID	3	Ref. 1
Seismic Effect	0.75 PSID		Ref. 1
Radiation Effect	0 PSID		Ref. 1
Humidity Effect	0 PSID		Ref. 1
Power Supply Effect	0.0296 (0.03) PSID	3	Ref. 1
RFI/EMI Effect	N/A		Ref. 1
Insulation Resistance Effect	N/A	3	Ref. 1
Over Pressure Effect	0 PSID		Ref. 1
Static Pressure Effect	2.161 PSID	3	Ref. 1

Plant data:	Value	Sigma if not 2	Reference(s)
Calib Temperature Range	65 – 90°F		Ref. 1
Normal Temperature Range	80 – 104°F		Ref. 1
Trip Temperature range	65 – 137°F		Ref. 1
Plant seismic value	4.2 gV 7 gH		Ref. 1
Plant Radiation value	0		Ref. 1
Plant Humidity value	50% AVERAGE		Ref. 2
Power Supply Variation value	±2 VDC		Ref. 1
RFI/EMI value	N/A		Ref. 1
Over Pressure value	N/A		Ref. 1
Static Pressure value	1025 PSI		Ref. 1

2.1 Component: Differential Pressure Transmitter (Cont.)

Drift:	Value	Sigma if not 2	Reference(s)
Current Calib. Interval	30 Months Vendor Data ☒ Includes 25%		Ref. 1
Desired Calib. Interval	30 Months Field Data Calc. ☒ Includes 25%		Ref. 2
Drift Source	☐ Vendor ☒ Calculated		Ref. 2
Drift Value (LOOP)	3.882 PSID (Overall Loop)		Ref. 2

Calibration:	Value / equation	Sigma if not 3	Reference(s)
As Left Tolerance (LOOP)	±0.85 PSID		Ref. 2
As Found Tolerance (LOOP)	±0.85 PSID		Ref. 2
Input Calibration Tool:	HEISE CMM		Ref. 2
Accuracy	See Appl Specific Input below		Ref. 1
Resolution / Readability	See Appl Specific Input below		Ref. 1
Minor Division	0.5 PSID		Ref. 1
Upper Range	400 PSID		Ref. 1
Temperature Effect	See Appl Specific Input below		Ref. 1
Input Calibration Standard:	See Appl Specific Input below		Ref. 1
Accuracy	See Appl Specific Input below		Ref. 1
Resolution / Readability	See Appl Specific Input below		Ref. 1
Minor Division	See Appl Specific Input below		Ref. 1
Upper Range	See Appl Specific Input below		Ref. 1
Temperature Effect	See Appl Specific Input below		Ref. 1
Output Calibration Tool:	FLUKE 8050A		Ref. 2
Accuracy	See Appl Specific Input below		Ref. 1
Resolution / Readability	See Appl Specific Input below		Ref. 1
Minor Division	See Appl Specific Input below		Ref. 1
Upper Range	See Appl Specific Input below		Ref. 1
Temperature Effect	See Appl Specific Input below		Ref. 1
Output Calibration Standard:	See Appl Specific Input below		Ref. 1
Accuracy	See Appl Specific Input below		Ref. 1
Resolution / Readability	See Appl Specific Input below		Ref. 1
Minor Division	See Appl Specific Input below		Ref. 1
Upper Range	See Appl Specific Input below		Ref. 1
Temperature Effect	See Appl Specific Input below		Ref. 1

Application Specific Input:	Value	Sigma if not 2	Reference(s)
Material Test Equipment Accuracy Overall Value (AMTE)	0.0014 x Span = 0.42 PSID	3	Ref. 1
NBS Traceable Equipment Accuracy	½ AMTE = 0.21 PSID	3	Ref. 1
MTE Readability (Analog only)	½ MTE Readability = 0.25	3	Ref. 1

2.2 Component: Trip Unit

Component Information:	Value/Equation	Reference(s)
Plant Instrument ID No.	1E31N0686A,B,C,D 1E31N0687A,B,C,D 1E31N0688A,B,C,D 1E31N0689A,B,C,D	Ref. 1
Instrument vendor	Rosemount	Ref. 1
Model ID No. (including Range Code)	510DU	Ref. 1
Plant Location(s)	Control Panel 1H13-P693	Ref. 1
Process Element	N/A	Ref. 1

Inputs:

Vendor specifications:	Value / Equation	Sigma if not 2	Reference(s)
Top of Scale	300 PSID		Comment 3
Bottom of Scale	0 PSID		Comment 3
Span	300 PSID		Ref. 1
Accuracy	SQR (((0.01/16) x span)^2 + (.002 x span)^2 + (.002 x span)^2)	3	Ref. 1
Temperature Effect	Included in Vendor Accuracy		Ref. 1
Seismic Effect	0 PSID		Ref. 1
Radiation Effect	0 PSID		Ref. 1
Humidity Effect	0 PSID		Ref. 1
Power Supply Effect	0 PSID		Ref. 1
RFI/EMI Effect	N/A		Ref. 1
Insulation Resistance Effect	N/A		Ref. 1
Over Pressure Effect	N/A		Ref. 1
Static Pressure Effect	N/A		Ref. 1

Plant data:	Value	Sigma if not 2	Reference(s)
Calib Temperature Range	N/R		Ref. 1
Normal Temperature Range	N/R		Ref. 1
Trip Temperature range	N/R		Ref. 1
Plant seismic value	N/A		Ref. 1
Plant Radiation value	N/A		Ref. 1
Plant Humidity value	N/A		Ref. 1
Power Supply Variation value	N/A		Ref. 1
RFI/EMI value	N/A		Ref. 1
Over Pressure value	N/A		Ref. 1
Static Pressure value	N/A		Ref. 1

2.2 Component: Trip Unit (Cont.)

Drift:	**Value**	**Sigma if not 2**	**Reference(s)**
Current Calib. Interval	30 Months Vendor Data ☒ Includes 25%		Ref. 2
Desired Calib. Interval	30 Months Field Data Calc. ☒ Includes 25%		Ref. 2
Drift Source	☐ Vendor ☒ Calculated		Ref. 2
Drift Value (LOOP)	Included with Transmitter Value, 3.882 PSID (Overall Loop)		Ref. 2

Calibration:	**Value / equation**	**Sigma if not 3**	**Reference(s)**
As Left Tolerance	Included with Transmitter Value, See Transmitter Data Sheet		Ref. 1, 2
As Found Tolerance	Included with Transmitter Value, See Transmitter Data Sheet		Ref. 1, 2
Input Calibration Tool:	Included with Transmitter Value, See Transmitter Data Sheet Appl Specific Input Section		Ref. 1
Accuracy			
Resolution / Readability			
Minor Division			
Upper Range			
Temperature Effect			
Input Calibration Standard:	Included with Transmitter Value, See Transmitter Data Sheet Appl Specific Input Section		Ref. 1
Accuracy			
Resolution / Readability			
Minor Division			
Upper Range			
Temperature Effect			
Output Calibration Tool:	See Transmitter Data Sheet Appl Specific Input Section		Ref. 1
Accuracy			
Resolution / Readability			
Minor Division			
Upper Range			
Temperature Effect			
Output Calibration Standard:	Included with Transmitter Value, See Transmitter Data Sheet Appl Specific Input Section		Ref. 1
Accuracy			
Resolution / Readability			
Minor Division			
Upper Range			
Temperature Effect			

3. Summary Results:

Calculated Values (Calculations Performed in Accordance with Reference 3)

Setpoint Function	Analytic Limit	Allowable Value	Setpoint	Meets LER Avoidance Criteria	Meets Spurious Trip Avoidance Criteria
MSL Isolation – Hi Flow	267.5 PSID	256.5 PSID	254 PSID	YES	YES

4. Comments and Recommendations:

1. The Primary Element Accuracy (PEA) has two parts, a random error and a bias error. The random error is due to the accuracy of the MSL venturi which is 2% of rated steam flow (Ref. 1) evaluated at a flow corresponding to the AL. Thus the random PEA error is:

   AL = 267.5 PSID (Ref 4)
   $PEA_{Random} = AL * \{(142/140)^2 - 1\}$
   $PEA_{Random} = 7.697$ PSID (2-SIGMA)

   The bias PEA error allows for condensate pot elevation differences of 0.5 PSID (Ref. 1)
   $PEA_{Bias} = 0.5$ PSIDc

2. The Process Measurement Accuracy (PMA) has only a bias part and is due to a 1% allowance for flow error due to pressure higher than the design pressure of the flow element (Ref. 5). The bias error is evaluated at a flow corresponding to the AL. Thus the random PMA error is:

   $PMA_{Bias} = AL * \{(141/140)^2 - 1\}$
   $PMA_{Bias} = 3.835$ PSID

3. The present instrument process range of –50 to 250 PSID (-49.375 to 246.875 corrected for static pressure span effect) must be recalibrated to encompass the new setpoint and Analytical Limit (AL). The new AL is 267.5 PSID with a recommended setpoint of 254.7 PSID (rounded slightly downward in the conservative direction to one figure after the decimal). The new recommended process calibrated range is 0 – 300 PSID. After correction for the static pressure span effect, the span remains the same at 296.25 PSID, but is now zero based.

5. References:

1. CEI Calculation, "Main Steam Line Isolation-High MSL Flow", calc. no. E31-C25, Rev.1, transmitted by item 1 of AIP–99–306, dated May 27,1999.

2. Data sheets transmitted by item 4 of AIP–99–306, dated May 27,1999, responding to GE-PAIP-340, 5/21/99 (Included in DRF).

3. "General Electric Methodology for Instrumentation Technical Specification and Setpoint Analysis," NEDC-32889P, Rev.2, Class 3, February 2000.

4. Analysis Completion Notification, Task G1-07 Nuclear Boiler System, Rev 0; DRF GE-NE-A22-00084-07-01

5. GE Setpoint Methodology NEDC – 31336P-A, Class 3, September 1996.

Perry Nuclear Power Plant Responses to an NRC Request For Additional Information (RAI) Forwarded by Reactor Systems Branch

The Perry Plant staff received a Request for Additional Information (RAI) from the NRC dated January 27, 2000. The RAI deals with questions associated with the Perry Plant license amendment request regarding a proposed increase of the present authorized rated thermal power level (power uprate) for the Perry Plant. The following are responses to requests from the NRC's Reactor Systems Branch.

NRC QUESTION

1. Topical report (Attachment 1 to the submittal) Section 4.3 states that ECCS performance was analyzed using NRC-approved SAFER/GESTR-LOCA methodology. When discussing ECCS performance evaluation methods, the Perry FSAR (Section 6.3.3) references NEDO-20566 (the GE generic LOCA analysis in accordance with Appendix K) but does not reference the SAFER/GESTR-LOCA topical report.

   The topical report also states that other safety analyses used the GEMINI transient analysis methods listed in NEDO-31897.

   a. Identify codes and methods used to obtain or confirm safety limits for the uprated power condition. Include the version and issue date for each item identified. Specifically list when SAFER/GESTR-LOCA was approved for use at Perry and when the associated plant-specific topical report was submitted to the NRC.

   b. Discuss any changes to the codes and methods identified in response to the above that were made since they were approved for use at Perry.

   c. Identify and discuss any limitations or conditions imposed upon approval of these methods for use at Perry.

RESPONSE

Prior to the power uprate evaluations, the Perry Plant implemented SAFER/GESTR methodology for pre-uprate conditions by updating the Updated Safety Analysis Report (USAR) pursuant to 10CFR 50.59 in April 1999. The update to USAR, Section 6.3.3, was included in Revision 10 of the Perry USAR effective October 1999. The reduction in peak cladding temperature due to the implementation of SAFER/GESTR was reported on June 7, 1999 by letter PY-CEI/NRR-2404L.

The transient analyses for the Perry Plant were done with the same set of methods as described in NEDC-31897. Updated versions of the principle codes would have been actually used in the analyses.

The following relates to power uprate evaluations:

a. For safety limit calculations, the following codes were used: PANAC10V, 06/06/96, and GESAM02V, 07/08/98. For stability, only PANACEA and ISCOR were used. PANAC10V on the VAX, issue date 8/16/96, and ISCOR09V on Alpha, issue date 11/6/97, were used.

   The transient analyses for the Perry Plant were done with the same set of methods as described in NEDC-31897. Updated versions of the principle codes would have been actually used in the analyses.

   For SAFER/GESTR-LOCA, the following codes/ versions were used to calculate the safety limits for the limiting GE11 fuel and to bound the GE10 and GE12 fuels.

   1. The LAMB model description is provided in Reference 1. The LAMB model has not been changed since Reference 1. LAMB-08A (October 1996) was used.
   2. The TASC (an improved SCAT model) description is provided in Reference 2 for GE11. The SCAT model description is provided in Reference 1. TASC-03V/A, Revision 1 (October 1991) was used.
   3. The GESTR-LOCA model is documented in Reference 3. The GESTR-LOCA model has not been changed since Reference 3. GESTR-08V was used.
   4. The SAFER model description is provided in References 4 and 7. The NRC acceptance of the SAFER/GESTR models is documented in References 5 and 6. SAFER04V was used.

b. For transient analyses, there are no changes in the methodology. For stability, there are no changes in the methodology. For SAFER/GESTR-LOCA, the SAFER code and application methodology have been changed since the NRC approval. The changes are documented in References 8 through 13.

c. There are no Perry Plant specific limitations and conditions that apply to the use of General Electric (GE) methods.

NRC QUESTION

2. Explain how the maximum extended operating domain and 100-percent rod lines were determined on the proposed power-to-flow map. A figure giving the power-to-flow map with both the current and proposed power scales would be helpful for comparison.

RESPONSE

The power uprate Maximum Extended Operating Domain (MEOD) line is just the pre-uprate MEOD line extended to 105% power and rescaled. The power uprate values (MWt vs. %Core Flow) below the original 100% MWt do not change. The pre-uprate 100% rod-line is the current 105% rod-line extended to 105% power and rescaled.

Figure 2-1, Perry 105% Power Uprate Two Loop Operation (TLO) Reactor Operating Domain from NEDE-32907P, "Safety Analysis Report for Perry 5% Thermal Power Uprate," (Attachment 1 of the power uprate license amendment request) is included (see page 8 of this RAI response). This Figure has a second Y-axis added to the left hand side in addition to the original (current) thermal power % on the same map.

NRC QUESTION

3. Provide power-to-flow maps showing the current stability control regions and the regions under power uprate conditions. Explain any differences with the interim corrective actions defined in GE SIL 380 and discussed NRC Bulletin 88-07 Supplement 1.

RESPONSE

Figure 2-1, Perry 105% Power Uprate TLO Reactor Operating Domain from NEDE-32907P (Attachment 1 of the power uprate license amendment request) is attached (see Page 8 of this RAI Response) with the Boiling Water Reactor Owners Group (BWROG) Interim Corrective Actions (ICA) regions illustrated. The Perry Plant modified operating procedures and operator training are consistent with, or more conservative than the BWROG guidelines as detailed in the response to Generic Letter 94-02, "Long-Term Solutions And Upgrade Of Interim Operating Recommendations For Thermal- Hydraulic Instabilities In Boiling Water Reactors", reference letter PY-CEI/NRR-1855L, dated September 4, 1994.

The regions have been rescaled to maintain the same absolute power and flow on the region boundaries as would exist for the revised ICAs prior to power uprate. In units of MWt, there is no change to the regions for power uprate conditions.

NRC QUESTION

4. The citation for Reference 10 in Section 4 of the topical report appears to be incorrect. Confirm that the reference should be NEDC-31984P rather than NEDO-30832A.

RESPONSE

Reference 10 in Section 4 of the topical report supports the following statement from Section 4.1.1.1 (b) of NEDC-32907P:

> "The local pool temperature limit for SRV discharge is specified in NUREG-0783, which was issued to address concerns regarding unstable condensation observed at high pool temperatures in plants without quenchers. Reference 10 provides justification for elimination of this limit for plants with quenchers on the SRV discharge lines."

NEDO-30832A is the correct reference. NEDC-31984P, Supplement 1, Section 3.8 reiterates that statement and cites NEDO-30832 as a reference.

NEDO-30832 was reissued as NEDO-30832A to include the NRC Safety Evaluation Report issued on August 29, 1994. NEDC-31984P could be used as the reference; however, the source reference for supporting the conclusion is NEDO-30832A.

NRC QUESTION

5. Attachment 6 to the submittal lists licensee commitments. Commitment number 9 states that safety evaluations are to be revised as necessary to include power uprate conditions. What licensee safety evaluations have been reviewed for suitability to uprated conditions, what safety evaluations have been revised, and what further safety evaluation reviews are planned?

RESPONSE

(1) NEDC-32907P, Section 11.1.2, plant-unique items and Section 11.1.2.1 lists the types of safety evaluations reviewed.

(2) None of the safety evaluations reviewed to date required revision due to the proposed 5% increase in Reactor Thermal Power.

(3) The reviews discussed in NEDC-32907P were completed in July 1999. New safety evaluations performed after July 1999 up until power uprate is implemented will be reviewed.

NRC QUESTION

6. Section 2.1 of the topical report states that parametric core design studies for Perry show that the power uprate can be accommodated. Describe the parametric studies and discuss the criteria used to judge that the results were acceptable for power uprate.

RESPONSE

Two core design studies were performed for the Perry Plant, which are described below:

1. A standard GE reload licensing analysis of the current Cycle 8 non-uprate Reference Loading Pattern at 105% uprate conditions (GE12 being the fresh fuel).

2. A fuel cycle analysis of Cycle 9 utilizing GE12 as the fresh fuel at 105% uprate conditions.

The criteria used to judge that these two core design results were acceptable for power uprate were respectively:

1. Compliance to GE's standard reload licensing analysis as described in GESTAR-II (NEDE-24011-P-A-13, August 1996). By showing compliance to GESTAR-II, this core design study showed that the 105% power uprate could be implemented for the current cycle through GE's standard reload licensing process once NRC approval is obtained for the 105% power uprate.

2. Satisfaction of reactor thermal and reactivity margins. By satisfying these margins, this core design study showed that the next cycle with GE12 could be licensed by GE's standard reload licensing process at 105% power uprate as a follow on to the Cycle 8 105% power uprate license.

NRC QUESTION

7. Summarize the sensitivity analyses discussed in Section 9.1 of the topical report that were conducted to determine the sensitivity of limiting transients to core flow, feedwater temperature, and cycle exposure. Include in the summary what events were considered, the ranges of input variables applied for each event considered, and what conclusions were drawn from the results.

RESPONSE

Table 9-1 of the topical report describes the core flow and temperature range for the transients evaluated for the Perry Plant. Table 9-2 describes all the transients evaluated and reports on the most limiting transient.

The input range of Feedwater (FW) flow is 75% flow to 105% flow. The input of the FW temperature of 420 °F was used for all transients except for the FW Controller Failure (FWCF), which was run at 250 °F. The increased core flow (105% flow) cases were more limiting than the low flow cases. For pressurization transients, only End Of Cycle (EOC) exposure cases were evaluated. The Loss of Feedwater Heating (LFWH) transient was evaluated at beginning of cycle, Middle Of Cycle (MOC), and EOC. The Rod Withdrawal Error event was evaluated at MOC.

NRC QUESTION

8. What analysis supports the statement in Section 9.2.3 of the topical report that systems used to respond to power restoration after a station blackout can restore suppression pool temperature to technical specification limits?

RESPONSE

Results of containment analyses demonstrated that suppression pool temperature remains below limiting conditions for Residual Heat Removal (RHR) Suppression Pool Cooling operation at the end of the 4-hour coping period. Results from this evaluation will be documented in a future revision to the USAR, Table 15H-1.

NRC QUESTION

9. What balance-of-plant modifications are associated with the power uprate?

RESPONSE

NEDC-32907P, Section 5.2.2, Electro Hydraulic Control (EHC) Turbine Control System, discusses Balance of Plant (BOP) modifications, which were necessary for power uprate.

No plant modifications are necessary to perform power uprate. However, Turbine first stage steam flow may limit operation to less than the full 5% power uprate . Therefore, it may be necessary to modify the main turbine by increasing the openings between the first stage turbine stationary blades to achieve the full 5% power uprate. This modification, if necessary, would be performed in an outage subsequent to power uprate implementation.

REFERENCES

References for SAFER/GESTR-LOCA and Responses to Questions 1a. and 1b. above:

1) "General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50 Appendix K", NEDO-20566A, General Electric Company, September 1986.

2) "GE11 Compliance with Amendment 22 of NEDE-24011-P-A (GESTAR-II)", NEDE-31917P, April 1991.

3) "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident, Volume I, GESTR-LOCA – A Model for the Prediction of Fuel Rod Thermal Performance", NEDC-23785-1-PA, General Electric Company, Revision 1, June 1984.

4) "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident, Volume III, SAFER/GESTR Application Methodology", NEDE-23785-1-PA, General Electric Company, revision 1, October 1984.

5) Letter, C.O. Thomas (NRC) to J.F. Quirk (GE), "Acceptance for Referencing of Licensing Topical Report NEDE-23785, Revision 1, Volume III (P), "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident", June 1, 1984.

6) "General Electric Standard Application for Reactor Fuel", NEDE-24011-P-A-14-US (GESTAR II), September 1999.

7) "SAFER Model for Evaluation of Loss-of-Coolant Accidents for Jet Pump and Non-Jet Pump Plants", NEDE-30966P-A, General Electric Company, October 1987.

8) MFN-040-88, H.C. Pfefferlen (GE) to J.A. Norberg (NRC), ECCS Evaluation Model Improvements, July 14, 1988.

9) MFN-023-90, R.C. Mitchell (GE) to USNRC, Reporting of Changes and Errors in ECCS Evaluation Models, June 13, 1990.

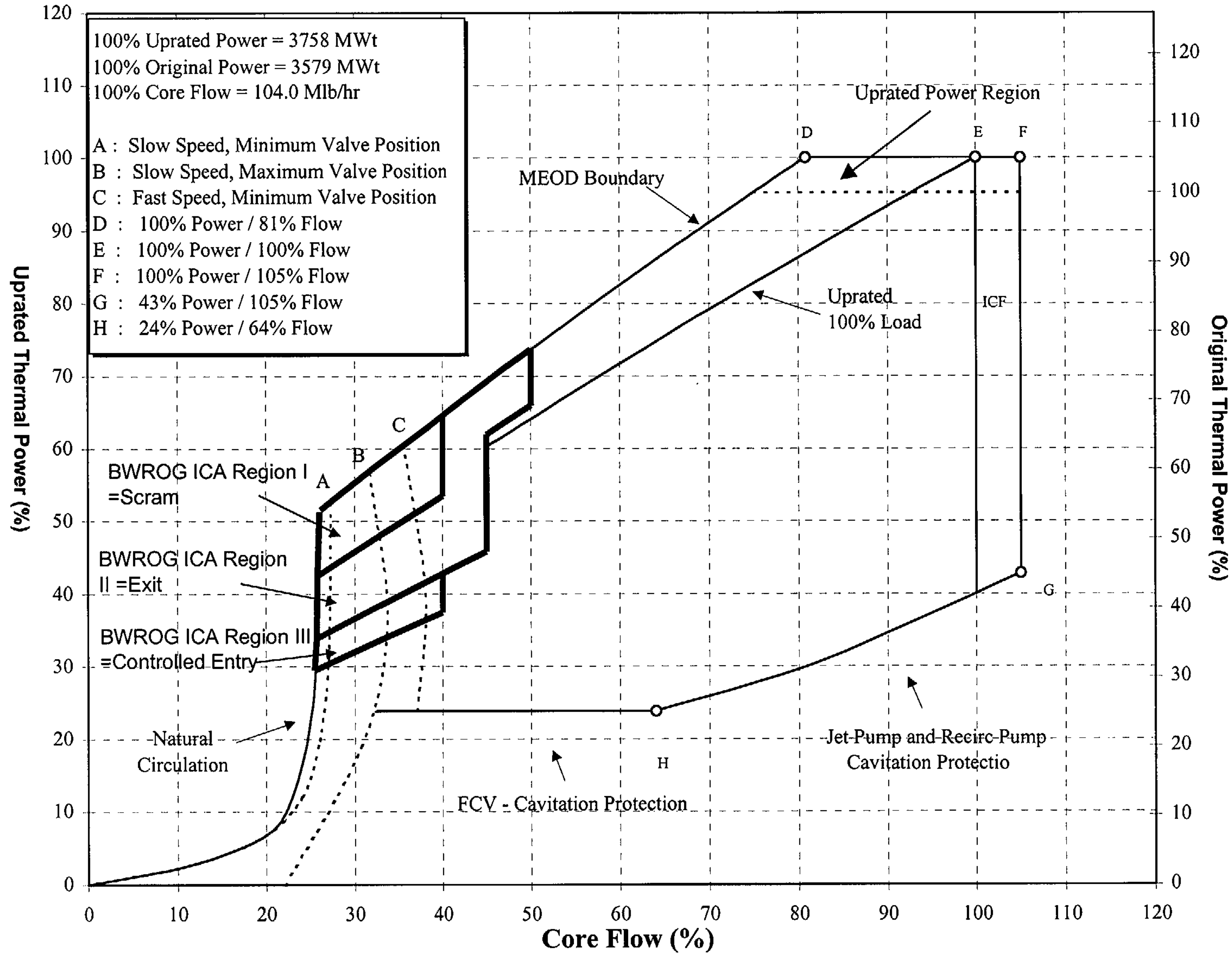
10) MFN-025-91, P.W. Marriott (GE) to USNRC, Reporting of Changes and Errors in ECCS Evaluation Models, March 12, 1991.

11) MFN-058-92, P.W. Marriott (GE) to USNRC, Reporting of Changes and Errors in ECCS Evaluation Models, June 26, 1992.

12) MFN-090-93, R.C. Mitchell (GE) to USNRC, Reporting of Changes and Errors in ECCS Evaluation Models, June 30, 1993.

13) MFN-020-96, R. J. Reda (GE) to USNRC, Reporting of Changes and Errors in ECCS Evaluation Models, February 20, 1996.

Perry 105% Power Uprate Two Loop Operation



CLARIFICATION REQUEST FROM FEBRUARY 8, 2000 CONFERENCE CALL

Within the Perry Plant power uprate submittal, Attachment 1, NEDC-32907P, "Safety Analysis Report for Perry 5% Thermal Power Uprate," the radiological consequences for the Perry Plant Design Basis Accidents (DBAs) were evaluated. Table 9-3 through Table 9-6 of NEDC-32907P lists the radiological consequences for the Perry Plant DBAs. The Total Effective Dose Equivalent (TEDE) values from the Revised Accident Source Term (RAST) methodology was used for the Loss of Coolant Accident (LOCA) dose consequences evaluation, Table 9-3. The original licensed whole body/thyroid dose considerations and not the RAST/TEDE values were used for the other events analyzed/evaluated (Main Steam Line Break Accident outside containment, Fuel Handling Accident, Control Rod Drop Accident, and Instrument Line Break Accident).

REVISED PROPOSED CHANGE TO

TECHNICAL SPECIFICATIONS

Attachment 4
PY-CEI/NRR-2470L
Page 2 of 2

Table 3.3.6.1-1 (page 1 of 6)
Primary Containment and Drywell Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Main Steam Line Isolation					
a. Reactor Vessel Water Level - Low Low Low, Level 1	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 14.3 inches
b. Main Steam Line Pressure - Low	1	2	E	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 795.2 psig
c. Main Steam Line Flow - High	1,2,3	2 per MSL	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ ~~180.3 psid~~ 256.5
d. Condenser Vacuum - Low	1,2(a), 3(a)	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≥ 7.6 inches Hg vacuum
e. Main Steam Line Pipe Tunnel Temperature - High	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.7	≤ 158.9°F
f. Main Steam Line Turbine Building Temperature-High	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 138.9°F
g. Manual Initiation	1,2,3	2	G	SR 3.3.6.1.5	NA
2. Primary Containment and Drywell Isolation					
a. Reactor Vessel Water Level - Low Low, Level 2	1,2,3	2(b)	H	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≥ -127.6 inches

(continued)

(a) With any turbine stop valve not closed.

(b) Required to initiate the associated drywell isolation function.