Entergy

Entergy Nuclear Northeast
Entergy Nuclear Operations, Inc.
Vermont Yankee
322 Governor Hunt Rd.
P.O. Box 157
Vernon, VT 05354
Tel 802-257-7711

October 28, 2003
BVY 03-98

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Subject: **Vermont Yankee Nuclear Power Station**
License No. DPR-28 (Docket No. 50-271)
Technical Specification Proposed Change No. 263 - Supplement No. 3
Extended Power Uprate – Updated Information

By letter dated September 10, 2003[1] and initially supplemented by letter dated October 1, 2003[2], Vermont Yankee[3] (VY) proposed to amend Facility Operating License, DPR-28, for the Vermont Yankee Nuclear Power Station (VYNPS) to increase the maximum authorized power level from 1593 megawatts thermal (MWt) to 1912 MWt. The letters dated September 10, 2003 and October 1, 2003, transmitted certain attachments which VY is updating herewith.

VY is providing three attachments to this letter: (1) an update to Attachment 3 of the September 10, 2003 submittal, which addresses VY's extended power uprate testing and modification plans; (2) an update to Attachment 7 of the September 10, 2003 submittal, which provides the justification for an exception to large transient testing; and (3) an update to Attachment 1 of the October 1, 2003 submittal, which is a review matrix that cross-references the criteria of NRC review standard RS-001[4] for extended power uprates with the information in the VYNPS Constant Pressure Power Uprate Safety Analysis Report[5] and the NRC-approved generic topical report for constant pressure power uprate[6]. Each of the attachments

[1] Vermont Yankee letter to U.S. Nuclear Regulatory Commission, "Extended Power Uprate," Proposed Change No. 263, BVY 03-80, September 10, 2003.
[2] Vermont Yankee letter to U.S. Nuclear Regulatory Commission, "Extended Power Uprate – Technical Review Guidance," Proposed Change No. 263, Supplement No. 1, BVY 03-90, October 1, 2003.
[3] Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. are the licensees of the Vermont Yankee Nuclear Power Station.
[4] U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, "Review Standard for Extended Power Uprates," RS-001 (Draft), December 2002.
[5] GE Nuclear Energy, "Safety Analysis Report for Vermont Yankee Nuclear Power Station Constant Pressure Power Uprate," NEDC-33090P, September 2003.
[6] GE Nuclear Energy, "Constant Pressure Power Uprate," Licensing Topical Report NEDC-33004P-A (proprietary), July 2003, and NEDO-33004-A (non-proprietary), July 2003.

A001

retains its previous numerical designation to minimize confusion and completely replaces the earlier version; thus, no other aspects of the license amendment request are affected.

If you have any questions with this submittal, please contact Mr. Len Gucwa at (802) 258-4225.

Sincerely,

Jay K. Thayer
Site Vice President

STATE OF VERMONT)
)ss
WINDHAM COUNTY)

Then personally appeared before me, Jay K. Thayer, who, being duly sworn, did state that he is Site Vice President of the Vermont Yankee Nuclear Power Station, that he is duly authorized to execute and file the foregoing document, and that the statements therein are true to the best of his knowledge and belief.

Sally A. Sandstrum, Notary Public
My Commission Expires February 10, 2007

SALLY A. SANDSTRUM
NOTARY PUBLIC
WINDHAM COUNTY, VERMONT

Attachments (3)
cc: (with attachments)

USNRC Region 1 Administrator
USNRC Resident Inspector – VYNPS
USNRC Project Manager – VYNPS (two copies)
Vermont Department of Public Service

Docket No. 50-271
BVY 03-98

Attachment

Vermont Yankee Nuclear Power Station

Technical Specification Proposed Change No. 263

Supplement No. 3

Extended Power Uprate – Updated Information

Modifications and Testing

EXTENDED POWER UPRATE

Modifications and Tests

The following is a list of currently planned modifications necessary to support extended power uprate (EPU) for Vermont Yankee Nuclear Power Station (VYNPS). These modifications will be implemented during the next two refueling outages (i.e., the scheduled refueling outages beginning in the Spring of 2004 (RFO-24) and Fall 2005 (RFO-25). The following modifications constitute planned actions on the part of Vermont Yankee. Further evaluations may identify the need for additional modifications or, on the contrary, obviate the need for some modifications. As such, this list is not a formal commitment to implement the modifications exactly as described or per the proposed schedule. Additionally, various minor modifications and adjustments to plant equipment, which may be necessary, are not listed.

VERMONT YANKEE CPPU MODIFICATIONS/TESTING		
Modification	**Description**	**Testing**
Main Turbine	Modifications include: • Replace HP Turbine steam path: • New control valve settings. • Modify control valve operating mechanism with 5% margin above CPPU conditions. • Modify turbine control and overspeed setpoint for CPPU conditions. • Replace 8th Stage diaphragms of the L.P. Turbine (Note: This modification will be implemented in RFO-25).	HP Turbine testing to include: • Overspeed testing. • Control Valve and Stop Valve testing. • As found and as left performance test.
Main Turbine Cross-around Relief Valve (CARV) Discharge Piping	Install higher capacity relief valves.	Relief valves to be bench tested prior to installation.
Main Generator System	Rewind/Upgrade the Main Generator for CPPU conditions. Replace bushing current transformers. (Note: bushing current transformers to be replaced in RFO-25).	Factory to perform applicable electrical testing of windings. Generator to be performance monitored.
Main Generator Cooling Hydrogen System	Replace Generator Hydrogen Coolers with upgraded coolers.	Performance monitoring.
Isolation Phase Bus Duct Cooling	Install a new Isolation Phase Bus Duct Cooling System to remove Bus Duct heat under CPPU conditions.	Performance monitoring.
HP Feedwater Heater Replacement	#1A, #1B, #2A, & #2B FW Heater Replacement.	Testing to include: • Pressure testing. • Visual Inspection. • Magnetic Particle testing. • Radiography. • In-service inspection. • Demonstration of thermal performance.

VERMONT YANKEE CPPU MODIFICATIONS/TESTING		
Modification	**Description**	**Testing**
Steam Dryer	Any identified modifications needed to maintain steam dryer structural integrity at CPPU conditions.	Performance Monitoring for Steam Dryer Cover Plates integrity includes: • Main Steam Line flow indication (check for unbalance). • RPV water level indication (check for unbalance). • Steam dome pressure (check for sudden drop). • Moisture carryover (unexpected step increase)
RHRSW System	Modify RHRSW Pumps (Train A & B) Motor Bearing Oil Coolers to recuperate Service Water flow to the coolers.	Testing of piping performed per modification to include: • Visual inspection. • Particle testing. • Ultrasonic flow testing. • In-Service inspection.
NSSS/BOP Instruments	Upgrade specific NSSS/BOP Instruments for CPPU conditions.	Perform instrument rescaling, calibration and functional testing.
NSSS/Torus Attached Piping	Upgrade particular NSSS and Torus attached piping supports.	As applicable, welds to be examined visual, liquid penetration and magnetic penetration methods.
Flow Induced Vibration (FIV)	Install/remove FIV Instrumentation.	Collect FIV background data and FIV CPPU data and analyze data.
Reactor Recirculation (RR) System	Permits continued reactor power operation by Recirculation Pumps speed running back to a preset demand if reactor is operating at or greater than a predetermined power level and one Feed Pump trips.	Modification testing to be performed with breakers in "test position" and RR System not operating.
Main Condenser	Tube stake Main Condenser tubing to reduce the effects of flow induced vibration.	Perform tube leak testing per the modification by Main Condenser flood-up.

VERMONT YANKEE CPPU MODIFICATIONS/TESTING		
Modification	**Description**	**Testing**
Condensate Demineralizer	Install a Condensate Demineralizer Filtered Bypass Strainer to permit one demineralizer to be removed under CPPU conditions.	With filtered bypass strainer in service, monitor flows under various CPPU conditions.
Feedwater System	Protect Feed Pumps with two sequential levels of low suction pressure trips at various time delays to ensure only one pump trips at a time.	Normal testing to be performed per modification testing, to be performed with breakers in "test position."
Cooling Tower Fans/Motors	Replace fan blades with more efficient blades and drive motors with upgraded higher performance motors.	Cooling Tower performance monitoring.
Core Spray & RHR Pump Seal Replacements (Contingency)	Core Spray and RHR pump seals may require replacement.	Leak check at pump rated conditions.
EQ Upgrades	Re-route feed to SRV monitor to new breaker.	Voltage check and meggar.

AGGREGATE IMPACT OF CPPU MODIFICATIONS TO DYNAMIC PLANT RESPONSE

The modifications listed on pages 2, 3, and 4 of this attachment were reviewed to ensure the aggregate impact of the modifications do not adversely impact the dynamic response of the plant to anticipated initiating events. This review has concluded that there is no adverse impact to the dynamic response of the plant to anticipated initiating events as a result of these plant modifications. A discussion of the modifications and their affect on integrated plant response is provided below.

All of the modifications listed in the "Vermont Yankee CPPU Modifications/Testing Table", with the exception of the Reactor Recirculation (RR) System, RHRSW System, Condensate Demineralizer, and Feedwater System, enhance and/or upgrade the existing plant components to allow for operation at CPPU conditions. With the exception of allowing operation at CPPU conditions, these modifications do not change the design functions of the equipment or the method of performing or controlling the function. Therefore, these modifications will not result in a significant change to the plant's dynamic response to anticipated initiating events.

The RR System modification provides an automatic runback of the recirculation pumps speed to a preset demand if the reactor is operating at or greater than a predetermined power level (~112% CLTP) and any one condensate or reactor feed pump trips. The runback lowers reactor power to be within the capability of the feedwater system with only two reactor feed pumps running so that reactor water level can be restored and maintained within the normal operating range. (Note: trip of a condensate pump at these power levels will result in the trip of a feedwater pump due to low feedwater pump suction pressure). Under current plant design, only two of the three reactor feedwater pumps are required for operation at CLTP. If one pump trips, a standby pump starts. If a standby pump is not available or does not start, operators would lower reactor power by reducing core flow. This modification turns a manual operator action, reducing core flow by lowering recirculation pumps speed, into an automatic action. The rate of the runback is within the current design of the control system. The plant dynamic response to a reactor feedwater pump trip is not significantly affected.

The RHRSW System modification reroutes the outlet of the RHRSW Pump Motor Bearing Oil Coolers so that the cooling water is directed back to the deep basin vice the current lineup which directs the outlet to a storm drain. This modification does not affect the cooling water flow rate to the motor bearing oil coolers and has no affect on any other system or component relied upon to mitigate anticipated initiating events. The plant dynamic response to any anticipated initiating events is not affected.

The Condensate Demineralizer modification installs a filtered bypass strainer to permit one demineralizer to be removed from service under CPPU conditions. Since feedwater system flow remains constant whether or not the filtered bypass strainer is in service, plant dynamic response to any anticipated initiating event is not affected.

The feedwater system modification provides two sequential levels of low suction pressure trips at various time delays to ensure only one reactor feed pump trips at a time. This modification improves the reliability of the feedwater system. Therefore, it presents no adverse affect on plant dynamic response to any anticipated initiating events.

CPPU POWER ASCENSION TEST PLAN																											
TEST	TEST DESCRIPTION	PRIOR TO STARTUP	PERCENT POWER CLTP (Allowance +0% -3%)																				(Allowance +0% -1%)				
			%	5	10	15	20	25	30	35	40	45	50	55	60	65	70	75	80	85	90	95	100	105	110	115	120
Main Turbine	Overspeed testing & backup overspeed testing.							X																			
Main Turbine	Demonstration of thermal performance improvements and generator increase.																										X
Main Turbine	Perform CPPU performance monitoring of the Main Turbine.							X					X					X					X	X	X	X	X
Flow Induced Vibration	Collect and analyze vibration data prior to power increases.												X					X					X	X	X	X	X
Condensate Demineralizer Filtered Bypass Strainer flow testing	Monitor the Condensate Demineralizer Filtered Bypass Strainer and demineralizer flows																						X	X	X	X	X
Core Thermal Limit Verification	Perform a Heat Balance Calculation.						X																X	X	X	X	X
Cooling Tower Modification	Perform CPPU performance monitoring.							X					X					X					X	X	X	X	X
Main Generator Modification	Perform CPPU performance monitoring.							X					X					X					X	X	X	X	X
Hydrogen Cooling Modification	Perform CPPU performance monitoring.							X					X					X					X	X	X	X	X
Iso Phase Bus Duct Modification	Perform CPPU performance monitoring.							X					X					X					X	X	X	X	X

CPPU POWER ASCENSION TEST PLAN																											
TEST	TEST DESCRIPTION	PRIOR TO STARTUP	PERCENT POWER CLTP (Allowance +0% -3%)																				(Allowance +0% -1%)				
			%	5	10	15	20	25	30	35	40	45	50	55	60	65	70	75	80	85	90	95	100	105	110	115	120
HP Feedwater Heater Modification	Perform demonstration of thermal performance.							X					X					X					X	X	X	X	X
Steam Dryer Modification	Monitor for steam dryer integrity.							X					X					X					X	X	X	X	X
Steam dryer/separator performance	Moisture carryover performance monitoring dryer separator integrity.																						X	X	X	X	X
BOP Monitoring	Performance monitoring of BOP Systems.							X					X					X					X	X	X	X	X
Radiation surveys.	Perform radiation surveys at various power levels.																						X	X	X	X	X
IRM Performance	IRM/APRM overlap will be done during the first controlled shutdown following APRM Calibration for EPU. If this is not possible, perform during next startup.																										
APRM Calibration	Calibrate each APRM channel to be consistent with the core thermal power, referenced to the LPU level, after the receipt of the CPPU SER from the NRC.																						X				

CPPU POWER ASCENSION TEST PLAN																											
TEST	TEST DESCRIPTION	PRIOR TO STARTUP	PERCENT POWER CLTP (Allowance +0% -3%)																				(Allowance +0% -1%)				
			%	5	10	15	20	25	30	35	40	45	50	55	60	65	70	75	80	85	90	95	100	105	110	115	120
Core Performance	Measure reactor and system parameters, calculate core thermal power and core performance parameters, evaluate data and project next power step's values.																				X		X	X	X	X	X
Pressure Regulator Tuning and Testing	Test and dynamically calibrate the pressure regulator system prior to start-up.	X																									
Pressure Regulator-Incremental Regulation Data Gathering	Average Main Steam line flow versus pressure regulator output. Data to be taken in less than 3% increments between 100% and 120% CLTP.																						X	XX	XX	XX	XX
Turbine First Stage Pressure	Validate the scram bypass function for the TSV Closure and TCV Fast Closure - Low Control Oil Pressure scram functions. Collect Data on the Turbine First Stage Pressure to Rated Thermal Power relationship over a band of 15%.					X	X	X	X																		
Pressure Regulator Testing: Bypass Valves	Pressure regulator control system response to a pressure setpoint change. This test to be performed on both pressure regulators.						X																				

CPPU POWER ASCENSION TEST PLAN

TEST	TEST DESCRIPTION	PRIOR TO STARTUP	PERCENT POWER CLTP (Allowance +0% -3%)																				(Allowance +0% -1%)				
			%	5	10	15	20	25	30	35	40	45	50	55	60	65	70	75	80	85	90	95	100	105	110	115	120
Pressure Regulator Testing: TCVs	Pressure regulator control system response to a pressure setpoint change. This test to be performed on both pressure regulators.																X				X		X	X	X	X	
Pressure Regulator Testing-Incremental Data Collection.	Collect Data from Generator minimum load to maximum CPPU power in <3% increments between 15% and 120% CLTP.					XX	XX	XX	XX	XX	XX	XX	XX		XX	XX	XX	XX	XX	XX	XX	XX	XX	XX	XX	XX	XX
Reactor Water Level Setpoint Changes	Test performed in single element and in three element control.																				X		X	X	X	X	
Manual Feedwater Flow Step Changes	Manually raise and lower feedwater flow using manual control.																				X		X	X	X	X	
Maximum Feedwater Runout Data Collection	Pressure, flow and controller data collected on feedwater system performance. Measured data is compared against expected values.																				X		X	X	X	X	

Comparison Of Initial Startup Testing And Planned CPPU Testing					
Test Number	Test Description Derived From VYNPS UFSAR Section 13.5 (Sub-Sections 13.5.2, 13.5.3 And 13.5.4)	Org. S/U Test Phase	AOO (System Challenge /Plant Transient) Test Aspect	Test Planned For CPPU (Yes/No)	Evaluation/Justification For Not Performing Test
STP 1	Chemical and Radiochemical: Chemical and radiochemical tests were conducted to establish water conditions prior to initial operations and to maintain these throughout the test program. Chemical and radiochemical checks were made at primary coolant, off-gas exhaust, waste, and auxiliary system sample locations. Base, or background, radioactivity levels were determined at this time for use in fuel assembly failure detection and long-range activity buildups studies.	Open Vessel Testing	None	Yes	Notes: 1,2, And 3
	Chemical and Radiochemical checks were made during heatup.	Initial Heatup	None		
	Chemical and Radiochemical Tests were continued.	Power Tests	None		
	Steam Separator-Dryer measurements of carryover and carryunder were made as a function of reactor water level and power level.	Power Tests	None		

Comparison Of Initial Startup Testing And Planned CPPU Testing					
Test Number	**Test Description Derived From VYNPS UFSAR Section 13.5 (Sub-Sections 13.5.2, 13.5.3 And 13.5.4)**	**Org. S/U Test Phase**	**AOO (System Challenge /Plant Transient) Test Aspect**	**Test Planned For CPPU (Yes/No)**	**Evaluation/Justification For Not Performing Test**
STP 2	Radiation Measurements were made for about a year prior to nuclear operation to establish base environmental monitoring levels.	Open Vessel Testing	None	Yes	Note 4
	Radiation Measurements were made periodically during heatup.	Initial Heatup	None		
	Radiation Measurements of limited extent were made at 25% of rated power and thorough surveys were made at 50%, 75%, and 100% power.	Power Tests	None		
STP 3	Fuel Loading: Fuel Loading was performed according to detailed, step-by-step written procedures. Control curtains were in place and the test and operational neutron sources were installed as required. Loading proceeded to the full size core.	Open Vessel Testing	None	No	The purpose of this test is to load fuel safely and efficiently to the full core size. Current Technical Specifications and approved plant procedures effectively govern the safe and efficient loading of fuel. No new fuel types are introduced for CPPU. CPPU has no affect on this test, therefore this test is not required.

Comparison Of Initial Startup Testing And Planned CPPU Testing					
Test Number	**Test Description Derived From VYNPS UFSAR Section 13.5 (Sub-Sections 13.5.2, 13.5.3 And 13.5.4)**	**Org. S/U Test Phase**	**AOO (System Challenge /Plant Transient) Test Aspect**	**Test Planned For CPPU (Yes/No)**	**Evaluation/Justification For Not Performing Test**
STP 4	Shutdown Margin: Shutdown Margin was demonstrated periodically during fuel loading that the reactor was subcritical by more than a specified amount with the single highest worth control rod withdrawn. The magnitude of the margin was chosen with consideration for expected reactivity changes during the first operating cycle and for the accuracy of measurement. The test had three parts: (A) The analytical determination of the control rod having the greatest reactivity worth, (B) The calibration of an adjacent control rod, determined analytically, and (C) The demonstration of subcriticality with the highest worth rod fully withdrawn and the second at the position needed to insert the required margin. This demonstration was made for the fully loaded core and for selected smaller core loadings.	Open Vessel Testing	None	No	The Shutdown Margin Requirement is not changed by CPPU. Shutdown Margin Testing is performed for each reload in accordance with approved plant procedures and Technical Specifications. This testing will be performed, as required by Technical Specifications, during the refueling outage that the CPPU core is loaded. Shutdown Margin Testing specifically for CPPU is not Required.

Comparison Of Initial Startup Testing And Planned CPPU Testing					
Test Number	Test Description Derived From VYNPS UFSAR Section 13.5 (Sub-Sections 13.5.2, 13.5.3 And 13.5.4)	Org. S/U Test Phase	AOO (System Challenge /Plant Transient) Test Aspect	Test Planned For CPPU (Yes/No)	Evaluation/Justification For Not Performing Test
STP 5	Control rod drive tests: Control rod drive system tests were performed on all drives prior to fuel loading to assure proper operability and to measure and adjust operating speeds. Functional testing of each drive was performed with dummy fuel just prior to fuel loading. Functional testing was again performed following the fuel loading in each cell. Drive pipe friction and scram times were determined for all drives at zero reactor pressure.	Open Vessel Testing	None	No	CRD scram time testing will be performed in accordance with Technical Specifications section 4.3.C and approved plant procedures. The performance of the CRD hydraulic system is independent of power level. There is no effect on the performance of the CRD system with no increase in reactor pressure. This is confirmed in section 2.5 of Attachment 4 to this LAR.
	Control rod drive system tests were made by measuring scram times on a selected number of drives at two intermediate pressures, scram times, and drive line friction tests on a representative set of drives at rated reactor pressure, and in-out driving times of selected rods during heatup.	Initial Heatup	None		

Comparison Of Initial Startup Testing And Planned CPPU Testing					
Test Number	**Test Description Derived From VYNPS UFSAR Section 13.5 (Sub-Sections 13.5.2, 13.5.3 And 13.5.4)**	**Org. S/U Test Phase**	**AOO (System Challenge /Plant Transient) Test Aspect**	**Test Planned For CPPU (Yes/No)**	**Evaluation/Justification For Not Performing Test**
STP 6	Control rod sequence: Control rod sequences were evaluated to verify that the stated criteria of safety, simplicity, and operating requirements are met during routine cold startup. The reactor was brought critical by withdrawing control rods in a specified sequence and reactivity addition rates are measured near critical. The pre-selected sequence was modified if necessary to meet criteria. A few nonstandard arrays were utilized to check out the operation of the rod worth minimizer. Control rod sequence to be used during the heatup was checked periodically for satisfactory performance.	Open Vessel Testing Initial Heatup	None None	No	This is an initial startup test requirement to achieve initial criticality in a safe and efficient manner for each of the two withdrawal sequences. Operation at CPPU increases the upper end of the power operating domain. These changes in the higher end do not significantly or directly affect the manner of operating or response of the reactor in the startup/low power range. Therefore, this test is not required. Plant startups will be performed using approved plant procedures.
STP 7	Calibration of rods: Calibrations of rods were performed to obtain reference relationships between control rod motion and reactor power and steam flow in the specified control rod sequence.	Power Tests	None	No	Operation at CPPU increases the upper end of the power operating domain. These changes in the higher end do not significantly or directly affect the manner of operating or response of the reactor in the startup/low power range. Therefore, this test is not required.
STP 8	Rod pattern exchange: Rod pattern exchanges were demonstrated from one specified control rod sequence to the other at the highest practical reactor power.	Power Tests	None	No	The methodology and approach taken to perform rod pattern exchange is not significantly affected by operation at CPPU conditions. Rod pattern exchanges will be performed in accordance with approved plant procedures and within fuel warranty requirements. This test is not required.

Comparison Of Initial Startup Testing And Planned CPPU Testing					
Test Number	**Test Description Derived From VYNPS UFSAR Section 13.5 (Sub-Sections 13.5.2, 13.5.3 And 13.5.4)**	**Org. S/U Test Phase**	**AOO (System Challenge /Plant Transient) Test Aspect**	**Test Planned For CPPU (Yes/No)**	**Evaluation/Justification For Not Performing Test**
STP 9	SRM Performance: Source Range Monitor (SRM) Performance. Adequate performance of the SRMs was established from data taken with the operational neutron sources in place. During initial reactor operations, the SRM subsystem was calibrated and its performance was compared with criteria on noise, signal-to-noise ratio, and response to change in core reactivity. SRM performance was determined by checking for proper overlap with the IRM subsystem.	Open Vessel Testing Initial Heatup	None None	No	SRM instrumentation is not modified for CPPU and operational neutron sources will not be required. Operation at CPPU increases the upper end of the power operating domain. These changes in the higher end do not significantly or directly affect the manner of operating or response of the reactor in the startup/low power range. Therefore, this test is not required.

Comparison Of Initial Startup Testing And Planned CPPU Testing					
Test Number	**Test Description Derived From VYNPS UFSAR Section 13.5 (Sub-Sections 13.5.2, 13.5.3 And 13.5.4)**	**Org. S/U Test Phase**	**AOO (System Challenge /Plant Transient) Test Aspect**	**Test Planned For CPPU (Yes/No)**	**Evaluation/Justification For Not Performing Test**
STP 10	Intermediate Range Monitor (IRM) Calibration: The IRMs were initially calibrated to give useful readings and to supply protection for this phase of the test program. This initial calibration was made by comparing to SRM readings in the overlap region. IRM Calibration. The IRM subsystem was recalibrated during heatup by making the IRM readings proportional to a known heat input to the reactor coolant from a non-nuclear heat source, such as the main recirculation pumps. The proportionality was determined by measuring the reactor coolant temperature rise produced by pump heating and by nuclear heating. IRM Calibration. The final calibration of the IRM system was made in the APRM-IRM power overlap region subsequent to the calibration of the APRM system.	Open Vessel Testing Initial Heatup Power Tests	None None None	Yes	Note 5
STP 11	LPRM calibrations, which included use of the traversing in-core probe (TIP) subsystem, were made at 50%, 75%, and 100% of rated power. Each local power range monitor was calibrated to read in terms of local fuel rod surface heat flux.	Power Tests	None	No	LPRM calibration is performed at a frequency specified in the Technical Specifications using approved plant procedures. The method and approach used to perform LPRM calibration is not affected by CPPU. This test is not required.

Comparison Of Initial Startup Testing And Planned CPPU Testing					
Test Number	**Test Description Derived From VYNPS UFSAR Section 13.5 (Sub-Sections 13.5.2, 13.5.3 And 13.5.4)**	**Org. S/U Test Phase**	**AOO (System Challenge /Plant Transient) Test Aspect**	**Test Planned For CPPU (Yes/No)**	**Evaluation/Justification For Not Performing Test**
STP 12	APRM Calibration: APRM calibrations were performed after making significant power level changes. Reactor heat balances formed the bases of the calibrations of these average power range monitors.	Power Tests	None	Yes	APRMs will be re-calibrated to read 100% at CPPU
STP 13	Process Computer: As station process variable signals became available to the computer, verification was made of these signals and of the computerized systems performance calculations. Process computer functions were verified as sensed variables came into range during the ascension to and at rated power.	Prior To Startup And During Open Vessel Testing Power Tests	None None	No	The plant process computer is maintained by approved plant procedures. Startup testing for this system is not required.
STP 14	RCIC: RCIC system was actuated when the reactor was shut down, but hot and pressurized, to demonstrate full capacity operation of the steam turbine driven pump.	Power Tests	See Evaluation/ Justification For Not Performing Test	No	RCIC Automatic Start From Cold Conditions (Performed At = 25% Power) – CLTR Section 3.9 indicates that there is no effect on the RCIC System for a Constant Pressure Power Uprate. This is confirmed in Section 3.9 of Att. 4 to this LAR. RCIC System testing, including automatic starts from cold conditions, is governed by Technical Specifications and approved plant procedures. As the CPPU does not have an effect on RCIC System, current surveillance testing remains valid for CPPU operations. Therefore, this testing is not required.

Comparison Of Initial Startup Testing And Planned CPPU Testing					
Test Number	**Test Description Derived From VYNPS UFSAR Section 13.5 (Sub-Sections 13.5.2, 13.5.3 And 13.5.4)**	**Org. S/U Test Phase**	**AOO (System Challenge /Plant Transient) Test Aspect**	**Test Planned For CPPU (Yes/No)**	**Evaluation/Justification For Not Performing Test**
STP 15	HPCI: HPCI system was tested to demonstrate proper performance of the system, including the steam turbine driven pump.	Power Tests	See Evaluation/ Justification For Not Performing Test	No	HPCI Automatic Start From Cold Conditions – CLTR Section 4.2 indicates that there is no effect on the HPCI System for a Constant Pressure Power Uprate. This is confirmed in Section 4.2 of Att. 4 to this LAR. HPCI System testing, including automatic starts from cold conditions, is governed by Technical Specifications and approved plant procedures. As the CPPU does not have an effect on the HPCI System, current surveillance testing remains valid for CPPU operations. Therefore, this testing is not required.
STP 16	Reactor Vessel Temperature: Reactor vessel temperatures were monitored during heatup and cooldown to check for proper operation and determine that specified temperature differences are not excessive.	Power Tests	None	No	This test obtains RPV temperatures during rapid heatup and cooldown to confirm thermal analysis models. CPPU does not affect the RPV temperatures during rapid heatup or cooldown. Since thermal analysis models were confirmed during the initial startup test subsequent testing is not required.
STP 17	System Expansion: System expansion checks were made during heatup to verify freedom of motion of major equipment and piping. System expansion tests were continued on a limited basis as reactor power was increased.	Initial Heatup Power Tests	None None	No	Since CPPU does not include a reactor vessel pressure increase, nor the corresponding primary coolant temperature increase, thermal expansion of drywell piping is not affected by CPPU. This test is not required.
STP 18	Power Distribution: Axial power distribution measurements were made with the tip system after significant changes in power, control rod pattern, or flow rate. The tip data were used for core performance evaluations and LPRM calibrations.	Power Tests	See Evaluation/ Justification For Not Performing Test	No	There are no changes to the tip system as a result of the CPPU. This test is not required.

Comparison Of Initial Startup Testing And Planned CPPU Testing					
Test Number	**Test Description Derived From VYNPS UFSAR Section 13.5 (Sub-Sections 13.5.2, 13.5.3 And 13.5.4)**	**Org. S/U Test Phase**	**AOO (System Challenge /Plant Transient) Test Aspect**	**Test Planned For CPPU (Yes/No)**	**Evaluation/Justification For Not Performing Test**
STP 19	Core Performance Evaluation: Core performance evaluations were made near or at rated temperature and pressure. This includes a reactor heat balance at rated temperature.	Initial Heatup	None	Yes	This test will be performed for CPPU
	Core performance evaluations were made periodically to demonstrate that the core was operating within allowable limits on maximum local surface heat flux and minimum critical heat flux ratio. This test included reactor heat balance determinations.	Power Tests	None		
STP 21	Flux Response To Rods: Flux response to control rod movements were determined in both equilibrium and transient conditions. Steady-state noise was measured if possible. Power-void loop stability was verified from this data.	Power Tests	None	No	This initial plant startup test was performed at 17% and 52% CLTP power. Operation at CPPU increases the upper end of the power operating domain. These changes in the higher end do not significantly or directly affect the manner of operating or response of the reactor at these lower power levels. Therefore, this test is not required.
STP 22	Pressure Regulator: Reactor pressure control was instituted using the main turbine pressure regulator.	Initial Heatup	See Evaluation	Yes	Setpoint Step Change and Simulated Failure Testing (Testing From 5% To 100% Power) – CLTR Section 5.2 indicates a CPPU effect on the reactor pressure control system due to the increased power level and steam flow. Section 10.4 of Att. 4 to this LAR requires testing to demonstrate acceptable CPPU performance.
	Pressure regulator tests were made to determine the response of the reactor and the turbine governor system. Regulator settings were optimized using data from this test.	Power Tests	See Evaluation		

Comparison Of Initial Startup Testing And Planned CPPU Testing					
Test Number	**Test Description Derived From VYNPS UFSAR Section 13.5 (Sub-Sections 13.5.2, 13.5.3 And 13.5.4)**	**Org. S/U Test Phase**	**AOO (System Challenge /Plant Transient) Test Aspect**	**Test Planned For CPPU (Yes/No)**	**Evaluation/Justification For Not Performing Test**
STP 23	Feedwater System: Feedwater pump trip tests were made to demonstrate reactor water level and plant response to loss of part of feedwater supply. Reactor water level changes were made to determine reactor response and to optimize level controller settings.	Power Tests	See Evaluation	Yes	Setpoint And Flow Change Testing (Testing From 15% To 100% Power) – CLTR Section 5.2 indicates a CPPU effect on the Feedwater Control System due to the increased power level and feedwater flow. Section 10.4 of Att. 4 to this LAR requires testing to demonstrate acceptable CPPU performance. Feedwater Pump Trip (Test Initial Conditions Range Between 75% And 100% Power) – CPPU LTR Section 9.1.3 indicates that loss of one feedwater pump has been included in CPPU transient analysis only for operational considerations (I.E., scram avoidance) and is not significantly affected by the CPPU. Therefore, this testing is not required. (Note 6)
STP 24	Bypass Valves: Bypass valve measurements were performed by opening a turbine bypass valve and recording the resulting reactor transients. Final adjustments to the pressure regulators were made.	Power Tests	See Evaluation	Yes	Bypass valve functional test (final test condition range between 85% and 100% power) – Power level at which valve testing can be performed without causing a scram / isolation may increase due to CPPU instrument-related changes (i.e., APRM re-calibration, MSL high steam flow isolation) as indicated in CLTR section 5.0. However, test power level is only a plant capacity consideration. Valve testing can continue to be performed at pre-CPPU power level (MWTh). Therefore, this testing is not required.

Comparison Of Initial Startup Testing And Planned CPPU Testing					
Test Number	**Test Description Derived From VYNPS UFSAR Section 13.5 (Sub-Sections 13.5.2, 13.5.3 And 13.5.4)**	**Org. S/U Test Phase**	**AOO (System Challenge /Plant Translent) Test Aspect**	**Test Planned For CPPU (Yes/No)**	**Evaluation/Justification For Not Performing Test**
STP 25	Main Steam Isolation Valves: Main Steam Isolation Valve functional tests were made at rated pressure. Main Steam Isolation Valve functional and operational tests were made as reactor power was increased.	Initial heatup Power tests	See Evaluation/ Justification For Not Performing Test	No	Single MSIV closure testing – Power level at which a single MSIV can be closed without a scram may increase due to CPPU instrument-related changes as indicated in CLTR section 5.0. However, this power level is a plant capacity consideration. Single MSIV closure can continue to be performed at pre-CPPU power level (MWTh). Therefore, this testing is not required. MSIV closure (test at 100% power) - This large transient test and others (i.e., generator load rejection, turbine trip) are evaluated for exemption from CPPU test program in Attachment 7 of this LAR.
STP 26	Relief Valves: This initial startup test was performed at ~19.5% CLTP to verify proper operation of the safety relief valves, verify proper sealing, and determine their capacity. Note: No test description for Relief Valve testing was identified in VYNPS UFSAR. Description is from GE Document No. 22A2217 Startup Test Specification for Vermont Yankee.	Power Tests	See Evaluation/ Justification For Not Performing Test	No	Technical Specifications and approved plant procedures govern the testing of the relief valves including manual opening of each relief valve once per cycle. Section 3.1 of Att. 4 to this LAR documents the acceptable evaluation of the overpressure protection and no effect on valve functionality by the CPPU. This test is not required.
STP 27	Turbine Trip: Turbine trip tests were performed to determine speed and reactor response.	Power Tests	See Evaluation/ Justification For Not Performing Test	No	This Large Transient Test and others (I.E., Generator Load Reject, MSIV Closure) are evaluated for exemption from CPPU test program in Attachment 7 of this LAR.

Comparison Of Initial Startup Testing And Planned CPPU Testing					
Test Number	**Test Description Derived From VYNPS UFSAR Section 13.5 (Sub-Sections 13.5.2, 13.5.3 And 13.5.4)**	**Org. S/U Test Phase**	**AOO (System Challenge /Plant Transient) Test Aspect**	**Test Planned For CPPU (Yes/No)**	**Evaluation/Justification For Not Performing Test**
STP 28	Generator Trip: Generator trip tests were performed to determine speed and reactor response.	Power Tests	See Evaluation/ Justification For Not Performing Test	No	This Large Transient Test And Others (I.E., Turbine Trip, MSIV Closure) are evaluated for exemption from CPPU test program in Attachment 7 of this LAR.
STP 29	Recirculation Flow Control: Flow Control capabilities were determined at specified power levels.	Power Tests	See Evaluation/ Justification For Not Performing Test	No	Flow Change Testing – CLTR Section 3.6 indicates a CPPU effect that increased voids in the core during normal uprate power operations requires a slight increase in recirculation drive flow to achieve the same core flow. Section 3.6 of Att. 4 to this LAR documents that the plant-specific system evaluation of the reactor recirculation system performance at CPPU power determines that adequate core flow can be maintained without requiring any changes to the recirculation system and only a small increase in pump speed for the same core flow, the response to flow changes will be similar to that of original startup testing. Therefore, this testing is not required.

Comparison Of Initial Startup Testing And Planned CPPU Testing					
Test Number	**Test Description Derived From VYNPS UFSAR Section 13.5 (Sub-Sections 13.5.2, 13.5.3 And 13.5.4)**	**Org. S/U Test Phase**	**AOO (System Challenge /Plant Transient) Test Aspect**	**Test Planned For CPPU (Yes/No)**	**Evaluation/Justification For Not Performing Test**
STP 30	Recirculation System: Recirculation pump trips and their effects on the jet pumps and the reactor were tested periodically during power increase.	Power Tests	See Evaluation/ Justification For Not Performing Test	No	**One Pump Trip** (Final Test At 100% Power) – CLTR Section 3.6 indicates a CPPU effect that increased voids in the core during normal uprate power operations requires a slight increase in recirculation drive flow to achieve the same core flow. Section 3.6 of Att. 4 to this LAR documents that the plant-specific system evaluation of the reactor recirculation system performance at CPPU power determines that adequate core flow can be maintained without requiring any changes to the recirculation system/pumps and only a small increase in their speed for the same core flow. The response to a one pump trip will be similar to that of original startup testing. Therefore, this testing is not required. **Two Pump Trip** (Final Test At 100% Power) – Section 3.6 of Att. 4 to this LAR indicates a CPPU effect that increased voids in the core during normal uprate power operations requires a slight increase in recirculation drive flow to achieve the same core flow. Section 3.6 of Att. 4 to this LAR documents that the plant-specific system evaluation of the reactor recirculation system performance at CPPU power determines that adequate core flow can be maintained without requiring any changes to the recirculation system/pumps and only a small increase in their speed for the same core flow. The response to a trip of both pumps will be similar to that of original startup testing. Therefore, this testing is not required.

Comparison Of Initial Startup Testing And Planned CPPU Testing					
Test Number	**Test Description Derived From VYNPS UFSAR Section 13.5 (Sub-Sections 13.5.2, 13.5.3 And 13.5.4)**	**Org. S/U Test Phase**	**AOO (System Challenge /Plant Transient) Test Aspect**	**Test Planned For CPPU (Yes/No)**	**Evaluation/Justification For Not Performing Test**
STP 31	Loss of Turbine-Generator and Off-Site Power: Auxiliary power loss tests were made to verify acceptable performance of the reactor and the electric equipment and auxiliary systems during the resulting transients.	Power Tests	See Evaluation/ Justification For Not Performing Test	No	Loss Of Turbine Generator and Off-Site Power Initial Test Condition was ~20% power – CLTR Section 6.1 indicates that, under emergency operations/distribution conditions (emergency diesel generators), the AC power supply and distribution components are considered adequate and an evaluation assures an adequate AC power supply to safety related systems. Section 6.2 of Att. 4 to this LAR documents the acceptable evaluation of the AC power system. Technical Specifications and approved plant procedures govern the testing of the safety related AC distribution system, including loss of off-site power tests. Operation at CPPU increases the upper end of the power operating domain and does not significantly or directly affect the manner of operating or response of the plant in the startup/low power range. Therefore, this test is not required.
STP 32	Recirculation M-G Set Speed Control: Flow control capabilities were determined at specified power levels.	Power Tests	None	No	This test determines the as built characteristics of the recirculation control system including the drive motor, fluid coupler, generator, drive pump and jet pumps. The CLTP recirculation pump speed range remains unchanged for operation at CPPU conditions. With the exception of adding a runback signal when a feedwater pump trips (Note 7) there are no modifications required to these components for CPPU. This test is not required since the recirculation system controls are unaffected by CPPU. (Note 8)

Comparison Of Initial Startup Testing And Planned CPPU Testing					
Test Number	**Test Description Derived From VYNPS UFSAR Section 13.5 (Sub-Sections 13.5.2, 13.5.3 And 13.5.4)**	**Org. S/U Test Phase**	**AOO (System Challenge /Plant Transient) Test Aspect**	**Test Planned For CPPU (Yes/No)**	**Evaluation/Justification For Not Performing Test**
STP X-5 (90)	Vibration Testing: Vibration measurements at cold flow conditions were performed as necessary to determine the vibrational characteristics of reactor vessel internals of Vermont Yankee design. The results of extensive vibration measurements made at other BWR installations were considered in selecting the components to be tested if it should be required. Vibration measurements were performed as necessary.	Open Vessel Test Power Tests	None None	No	This test obtains vibration measurements on various reactor pressure vessel internals to demonstrate the mechanical integrity of the system under conditions of flow induced vibration, and to check the validity of the analytical vibration model. Analysis of the reactor vessel internals at CPPU power level was performed to ensure that the design continues to comply with the existing structural requirements. (Note 9)

Notes to Table

1. For CPPU Testing, added demonstration of proper steam separator-dryer operation.
2. Startup test included objective to determine that the sampling equipment, procedures and analytical techniques are adequate to supply the data required to demonstrate that the coolant chemistry meets water quality specifications and process requirements. This objective is not applicable to CPPU and is not required.
3. Startup test included objective to evaluate the performance of the fuel, operation of the demineralizers and filters, condenser integrity, operation of the off gas system and calibration of certain process instruments. The current Vermont Yankee chemistry and plant performance monitoring programs gather information on plant equipment and system performance. This information is evaluated in order to maintain equipment, system and plant performance within process requirements, chemistry/radiochemistry specifications and guidelines and fuel warrantee. This testing is not required for CPPU implementation.
4. Startup test included objective to determine the background radiation levels in the plant environs prior to operation for base data on activity buildup. This initial startup requirement is not applicable to CPPU and is not required.
5. The IRM overlap with the SRMs is not affected by CPPU. The APRMs will be re-referenced to read 100% at CPPU conditions, therefore, the IRM performance test will be performed to reestablish the IRM to APRM overlap.
6. Feedwater System startup testing included a feedwater pump trip test. For this test one of two operating feedwater pumps was tripped and the standby feedwater pump was allowed to automatically start. At CPPU conditions all three feedwater pumps will be required; there will be no standby pump available. This test is not required for CPPU.
7. Testing associated with the recirculation pump runback on reactor feed pump trip is discussed above in the modification testing section of this attachment.
8. The recirculation system will have to overcome a slight increase in two-phase flow resistance due to an increase in the core average void fraction. The system will accommodate the expected insignificant increase at CPPU condition when operating at maximum core flow.
9. Results of this analysis are in section 3.4.2 of NEDC-33090P (Attachment 4 of the LAR).

Docket No. 50-271
BVY 03-98

Attachment

Vermont Yankee Nuclear Power Station

Technical Specification Proposed Change No. 263

Supplement No. 3

Extended Power Uprate – Updated Information

Justification for Exception to Large Transient Testing

JUSTIFICATION FOR EXCEPTION TO LARGE TRANSIENT TESTING

Background

The basis for the Constant Pressure Power Uprate (CPPU) request was prepared following the guidelines contained in the NRC approved, General Electric (GE) Company Licensing Topical Report for Constant Pressure Power Uprate (CLTR) Safety Analysis: NEDC-33004P-A Rev. 4, July 2003. The NRC staff did not accept GEs proposal for the generic elimination of large transient testing (i.e., Main Steam Isolation Valve (MSIV) closure and turbine generator load rejection) presented in NEDC-33004P Rev. 3. Therefore, on a plant specific basis, Vermont Yankee Nuclear Power Station (VYNPS) is taking exception to performing the large transient tests; MSIV closure, turbine trip, and generator load rejection.

The CPPU methodology, maintaining a constant pressure, simplifies the analyses and plant changes required to achieve uprated conditions. Although no plants have implemented an Extended Power Uprate (EPU) using the CLTR, thirteen plants have implemented EPUs without increasing reactor pressure.

- Hatch Units 1 and 2 (105% to 113% of Original Licensed Thermal Power (OLTP))
- Monticello (106% OLTP)
- Muehleberg (i.e., KKM) (105% to 116% OLTP)
- Leibstadt (i.e., KKL) (105% to 117% OLTP)
- Duane Arnold (105% to 120% OLTP)
- Brunswick Units 1 and 2 (105% to 120% OLTP)
- Quad Cities Units 1 and 2 (100% to 117% OLTP)
- Dresden Units 2 and 3 (100% to 117% OLTP)
- Clinton (100% to 120%)

Data collected from testing responses to unplanned transients for Hatch Units 1 and 2 and KKL plants has shown that plant response has consistently been within expected parameters.

Entergy believes that additional MSIV closure, turbine trip, and generator load rejection tests are not necessary. If performed, these tests would not confirm any new or significant aspect of performance that is not routinely demonstrated by component level testing. This is further supported by industry experience which has demonstrated plant performance, as predicted, under EPU conditions. VYNPS has experienced generator load rejections from 100% current licensed thermal power (see VYNPS Licensee Event Reports (LER) 91-005, 91-009, and 91-014). No significant anomalies were seen in the plant's response to these events. Further testing is not necessary to demonstrate safe operation of the plant at CPPU conditions. A Scram from high power level results in an unnecessary and undesirable transient cycle on the primary system. In addition, the risk posed by intentionally initiating a MSIV closure transient, a turbine trip, or a generator load rejection, although small, should not be incurred unnecessarily.

VYNPS Response to Unplanned Transients:

VYNPS experienced an unplanned Generator Load Rejection from 100% power on 04/23/91. The event included a loss of off site power. A reactor scram occurred as a result of a turbine/generator trip on generator load rejection due to the receipt of a 345 KV breaker failure signal. This was reported to the NRC in LER 91-009, dated 05/23/91. No significant anomalies were seen in the plant's response to this event. VYNPS also experienced the following unplanned generator load rejection events:

- On 3/13/91 with reactor power at 100% a reactor scram occurred as a result of turbine/generator trip on generator load rejection due to a 345KV Switchyard Tie Line Differential Fault. This event was reported to the NRC in LER 91-005, dated 4/12/91.
- On 6/15/91 during normal operation with reactor power at 100% a reactor scram occurred due to a Turbine Control Valve Fast Closure on Generator Load Rejection resulting from a loss of the 345KV North Switchyard bus. This event was reported to the NRC in LER 91-014, dated 7/15/91.

No significant anomalies were seen in the plant's response to these events. Transient experience at high powers and for a wide range of power levels at operating BWR plants has shown a close correlation of the plant transient data to the predicated response.

Based on the similarity of plants, past transient testing, past analyses, and the evaluation of test results, the effects of the CPPU RTP level can be analytically determined on a plant specific basis. The transient analysis performed for the VYNPS CPPU demonstrates that all safety criteria are met and that this uprate does not cause any previous non-limiting events to become limiting. No safety related systems were significantly modified for the CPPU, however some instrument setpoints were changed. The instrument setpoints that were changed do not contribute to the response to large transient events. No physical modification or setpoint changes were made to the SRVs. No new systems or features were installed for mitigation of rapid pressurization anticipated operational occurrences for this CPPU. A Scram from high power level results in an unnecessary and undesirable transient cycle on the primary system. Therefore, additional transient testing involving scram from high power levels is not justifiable. Should any future large transients occur, VYNPS procedures require verification that the actual plant response is in accordance with the predicted response. Existing plant event data recorders are capable of acquiring the necessary data to confirm the actual versus expected response.

Further, the important nuclear characteristics required for transient analysis are confirmed by the steady state physics testing. Transient mitigation capability is demonstrated by other equipment surveillance tests required by the Technical Specifications. In addition, the limiting transient analyses are included as part of the reload licensing analysis.

MSIV Closure Event

Closure of all MSIVs is an Abnormal Operational Transient as described in Chapter 14 of the VYNPS Updated Final Safety Analysis Report (UFSAR). The transient produced by the fast closure (3.0 seconds) of all main steam line isolation valves represents the most severe abnormal operational transient resulting in a nuclear system pressure rise when direct scrams are ignored. The Code overpressure protection analysis assumes the failure of the direct isolation valve position scram. The MSIV closure transient, assuming the backup flux scram verses the valve position scram, is more significant. This case has been re-evaluated for CPPU with acceptable results.

The CLTR states that: "The same performance criteria will be used as in the original power ascension tests, unless they have been replaced by updated criteria since the initial test program." The original MSIV closure test allowed the scram to be initiated by the MSIV position switches. As such, if the original MSIV closure test were re-performed, the results would be much less significant than the MSIV closure analysis performed by GE for CPPU.

The original MSIV closure test was intended to demonstrate the following:

1. *Determine reactor transient behavior during and following simultaneous full closure of all MSIVs.*

   Criteria:
   a) Reactor pressure shall be maintained below 1230 psig.
   b) Maximum reactor pressure should be 35 psi below the first safety valve setpoint. (This is margin for safety valve weeping).

2. *Functionally check the MSIVs for proper operation and determine MSIV closure time.*

   Criteria:
   a) Closure time between 3 and 5 seconds.

Item 1: Reactor Transient Behavior

For this event, the closure of the MSIVs cause a vessel pressure increase and an increase in reactivity. The negative reactivity of the scram from MSIV position switches should offset the positive reactivity of the pressure increase such that there is a minimal increase in heat flux. Therefore, the thermal performance during the proposed MSIV closure test is much less limiting than any of the transients routinely re-evaluated. CPPU will have minimal impact on the components important to achieving the desired thermal performance. Reactor Protection system (RPS) logic is unaffected and with no steam dome pressure increase, overall control rod insertion times will not be significantly affected. MSIV closure speed is controlled by adjustments to the actuator and is considered very reliable as indicated below.

Reactor Pressure

Due to the minimal nature of the flux transient, the expected reactor pressure rise, Item 1 above, is largely dependent on SRV setpoint performance. At VYNPS all four SRVs are replaced with re-furbished and pre-tested valves each outage. After the outage, the removed valves are sent out for testing and recalibration for installation in the following outage. Over the past ten years there have been twenty five SRV tests performed. In those twenty five tests only one test found the as-found setting outside the Technical Specification (TS) current allowable tolerance of ±3%. This valve was found to deviate by 3.4% of its nominal lift setpoint. Note that this is bounded by the VYNPS design analysis for peak vessel pressure which assumes one of the four SRVs does not open at all (one SRV out of service). Given the historical performance of the VYNPS SRVs along with the design margins, performance of an actual MSIV closure test would provide little benefit for demonstrating vessel overpressure protection that is not already accomplished by the component level testing that is routinely performed, in accordance with the VYNPS TSs.

Because rated vessel steam dome pressure is not being increased and SRV setpoints are not being changed, there is no increase in the probability of leakage after a SRV lift. Since SRV leakage performance is considered acceptable at the current conditions, which match CPPU conditions with respect to steam dome pressure and SRV setpoints, SRV leakage performance should continue to be acceptable at CPPU conditions. An MSIV closure test would provide no significant additional confirmation of Item 1 performance criteria than the routine component testing performed every cycle, in accordance with the VYNPS TSs.

Item 2: MSIV Closure Time

Since steam flow assists MSIV closure, the focus of Item 2 was to verify that the steam flow from the reactor was not shut off faster than assumed (i.e., 3 seconds). During maintenance and surveillance, MSIV actuators are evaluated and adjusted as necessary to control closure speed, and VYNPS test performance has been good. To account for minor variations in stroke times, the calibration test procedure for MSIV closure (OP 5303) requires an as left fast closure time of 4.0 $\pm$0.2 seconds. The MSIVs were evaluated for CPPU. The evaluation included MSIV closure time and determined that the MSIVs are acceptable for CPPU operation. Industry experience, including VYNPS, has shown that there are no significant generic problems with actuator design. Confidence is very high that steam line closure would not be less than assumed by the analysis.

Other Plant Systems and Components Response

The MSIV limit switches that provide the scram signal are highly reliable devices that are suitable for all aspects of this application including environmental requirements. There is no direct effect by any CPPU changes on these switches. There may be an indirect impact caused by slightly higher ambient temperatures, but the increased temperatures will still be below the qualification temperature. These switches are expected to be equally reliable before and after CPPU.

The Reactor Protection System (RPS) and Control Rod Drive (CRD) components that convert the scram signals into CRD motion are not directly affected by any CPPU changes. Minor changes in pressure drops across vessel components may result in very slight changes in control blade insertion rates. These changes have been evaluated and determined to be insignificant. The ability to meet the scram performance requirement is not affected by CPPU. Technical Specification (TS) requirements for these components will continue to be met.

CPPU Modifications

Feedwater System operation will require operation of all three feed pumps at CPPU conditions (unlike CLTP conditions). Operation of the additional Reactor Feed Pump (RFP) will not affect plant response to an MSIV closure transient. All feedwater pumps receive a trip signal prior to level reaching 177 inches. Overfill of the vessel after a trip would only occur if level exceeded approximately 235.5 inches. Since the feedwater pumps, the High Pressure Coolant Injection (HPCI) turbine, and the Reactor Core Isolation Cooling (RCIC) turbine all receive trip signals prior to level reaching 177 inches, a substantial margin exists. VYNPS operating history has demonstrated that this margin greatly exceeds vessel level overshoot during transient events. Based on this, there is adequate confidence that the vessel level will remain well below the main steam lines under CPPU conditions. The HPCI and RCIC pump trip functions are routinely verified as required by TSs and are considered very reliable.

The modification adding a recirculation pump runback following a RFP trip will not affect the plant response to this transient. The reactor scram signal from the MSIV limit switches will result in control rod insertion prior to any manual or automatic operation of the RFPs. Since control rods will already be inserted, a subsequent runback of the recirculation pumps will not affect the plant response.

The modification (BVY 03-23 "ARTS/MELLLA") to add an additional unpiped Spring Safety Valve (SSV) will not affect the plant response to this transient. The new third SSV will have the same lift setpoint as the two existing SSVs. This transient does not result in an opening of a SSV, nor is credit taken for SSV actuation.

Generator Load Reject and Turbine Trip Testing

"Generator Load Rejection From High Power Without Bypass" (GLRWB) is an Abnormal Operational Transient as described in Chapter 14 of the VYNPS Updated Final Safety Analysis Report (UFSAR). This transient competes with the turbine trip without bypass as the most limiting overpressurization transient that challenges thermal limits for each cycle. The turbine trip and generator load reject are essentially interchangeable. The only differences are 1) whether the RPS signal originates from the acceleration relay (GLRWB) or from the main turbine stop valves (turbine trip), and 2) whether the control valves close shutting off steam to the turbine or the stop valves close to isolate steam to the turbine. Both tests would verify the same analytical model for plant response. Therefore, the GLRWB is considered bounding or equivalent to the Turbine Trip.

The GLRWB analysis assumes that the transient is initiated by a rapid closure of the turbine control valves. It also assumes that all bypass valves fail to open. The CLTR states that: "The same performance criteria will be used as in the original power ascension tests, unless they have been replaced by updated criteria since the initial test program." The startup test for generator load reject allowed the select rod insert feature to reduce the reactor power level and, in conjunction with bypass valve opening, control the transient such that the reactor does not scram. Current VYNPS design does not include the select rod insert feature. The plant was also modified to include a scram from the acceleration relay of the turbine control system. Under current plant design, the original generator load reject test can not be re-performed. If a generator load reject with bypass test were performed, the results would be much less significant than the generator load reject without bypass closure analysis performed for CPPU.

The original generator load reject test was intended to demonstrate the following:

1. *Determine and demonstrate reactor response to a generator trip, with particular attention to the rates of changes and peak values of power level, reactor steam pressure and turbine speed.*

   Criteria:
   a. *All test pressure transients must have maximum pressure values below 1230 psig*
   b. *Maximum reactor pressure should be 35 psi below the first safety valve setpoint. (This is margin for safety valve weeping).*
   c. *The select rod insert feature shall operate and in conjunction with proper bypass valve opening, shall control the transient such that the reactor does not scram.*

Due to plant modification discussed above, criterion c. above would no longer be applicable for a generator load reject test. The generator load reject startup test was performed at 93.7% power; however, a reactor scram occurred during testing and invalidated the test. A design change to initiate an immediate scram on generator load reject was implemented and this startup test was subsequently cancelled since it was no longer applicable.

Item 1 Reactor Response

For a generator load reject with bypass event, given current plant design, the fast closure of the Turbine Control Valves (TCVs) cause a trip of the acceleration relay in the turbine control system. The acceleration relay trip initiates a full reactor scram. The bypass valves open, however, since the capacity of the bypass valves at CPPU is 87%, vessel pressure increases. This results in an increase in reactivity. The negative reactivity of the TCV fast closure scram from the acceleration relay should offset the positive reactivity of the pressure increase such that there is a minimal increase in heat flux. Therefore, the thermal performance during a generator load rejection test would be much less limiting than any of the transients routinely re-evaluated. CPPU will have minimal impact on the components important to achieving the desired thermal performance. Reactor Protection system (RPS) logic is unaffected and with no steam dome pressure increase, overall control rod insertion times will not be significantly affected. A trip channel and alarm functional test of the turbine control valve fast closure scram is performed every three months in accordance with plant technical specifications. This trip function is considered very reliable.

Reactor Pressure

Due to the minimal nature of the flux transient, the expected reactor pressure rise, Criteria a. and b. above, are largely dependent on SRV setpoint performance. Refer to the MSIV closure Reactor Pressure section above for discussion of SRV setpoint performance.

Because rated vessel steam dome pressure is not being increased and SRV setpoints are not being changed, there is no increase in the probability of leakage after a SRV lift. Since SRV leakage performance is considered acceptable at the current conditions, which match CPPU conditions with respect to steam dome pressure and SRV setpoints, SRV leakage performance will continue to be acceptable at CPPU conditions. A generator load rejection test would provide no significant additional confirmation of performance criteria a. and b. than the routine component testing performed every cycle, in accordance with the VYNPS TSs.

Other Plant Systems and Components Response

The turbine control system acceleration relay hydraulic fluid pressure switches that provide the scram signal are highly reliable devices that are suitable for all aspects of this application including environmental requirements. There is no direct effect by any CPPU changes on these pressure switches. These switches are expected to be equally reliable before and after CPPU.

The Reactor Protection System (RPS) and Control Rod Drive (CRD) components that convert the scram signals into CRD motion are not directly affected by any CPPU changes. Minor changes in pressure drops across vessel components may result in very slight changes in control blade insertion rates. These changes have been evaluated and determined to be insignificant. The ability to meet the scram performance requirement is not affected by CPPU. TS requirements for these components will continue to be met.

CPPU Modifications

As previously described, Feedwater System operation will require all three feed pumps at CPPU conditions. Operation of the additional Reactor Feed Pump (RFP) will not affect plant response to this transient. All feedwater pumps receive a trip signal prior to level reaching 177 inches. Overfill of the vessel after a trip would only occur if level exceeded approximately 235.5 inches. Since the feedwater pumps, the High Pressure Coolant Injection (HPCI) turbine, and the RCIC turbine all receive trip signals prior to level reaching 177 inches, a substantial margin exists. VYNPS operating history has demonstrated that this margin greatly exceeds vessel level overshoot during transient events. Based on this, there is adequate confidence that the vessel level will remain well below the main steam lines under CPPU conditions. The HPCI and RCIC pump trip functions are routinely verified as required by TSs and are considered very reliable.

The modification adding a recirculation pump runback following a RFP trip will not affect the plant response to this transient. The reactor scram signal from turbine control valve fast closure will result in control blade insertion prior to any manual or automatic operation of the RFPs. Since control blades will already be inserted, a subsequent runback of the recirculation pumps will not affect the plant response.

The ARTS/MELLLA modification (BVY 03-23) to add an additional unpiped SSV will not affect the plant response to this transient. The new third SSV will have the same lift setpoint of the two existing SSVs. This transient does not result in an opening of a SSV nor is credit taken for SSV actuation.

HP Turbine modification replaces the steam flow path but will not affect the turbine control system hydraulic pressure switches that provide the turbine control valve fast closure scram signal to the RPS system.

Industry Boiling Water Reactor (BWR) Power Uprate Experience

Southern Nuclear Operating Company's (SNC) application for EPU of Hatch Units 1 and 2 was granted without requirements to perform large transient testing. VYNPS and Hatch are both BWR/4 with Mark 1 containments. Although Hatch was not required to perform large transient testing, Hatch Unit 2 experienced an unplanned event that resulted in a generator load reject from 98% of uprated power in the summer of 1999. As noted in SNOC's LER 1999-005, no anomalies were seen in the plant's response to this event. In addition, Hatch Unit 1 has experienced one turbine trip and one generator load reject event subsequent to its uprate (i.e., LERs 2000-004 and 2001-002). Again, the behavior of the primary safety systems was as expected. No new plant behaviors were observed that would indicate that the analytical models being used are not capable of modeling plant behavior at EPU conditions.

The KKL power uprate implementation program was performed during the period from 1995 to 2000. Power was raised in steps from its previous operating power level of 3138 MWt (i.e., 104.2% of OLTP) to 3515 MWt (i.e., 116.7% OLTP). Uprate testing was performed at 3327 MWt (i.e., 110.5% OLTP) in 1998, 3420 MWt (i.e., 113.5% OLTP) in 1999 and 3515 MWt in 2000.

KKL testing for major transients involved turbine trips at 110.5% OLTP and 113.5% OLTP and a generator load rejection test at 104.2% OLTP. The KKL turbine and generator trip testing demonstrated the performance of equipment that was modified in preparation for the higher power levels. Equipment that was not modified performed as before. The reactor vessel pressure was controlled at the same operating point for all of the uprated power conditions. No unexpected performance was observed except in the fine-tuning of the turbine bypass opening that was done as the series of tests progressed. These large transient tests at KKL demonstrated the response of the equipment and the reactor response. The close matches observed with predicted response provide additional confidence that the uprate licensing analyses consistently reflected the behavior of the plant.

Plant Modeling, Data Collection, and Analyses

From the power uprate experience discussed above, it can be concluded that large transients, either planned or unplanned, have not provided any significant new information about transient modeling or actual plant response. Since the VYNPS uprate does not involve reactor pressure changes, this experience is considered applicable.

The safety analyses performed for VYNPS used the NRC-approved ODYN transient modeling code. The NRC accepts this code for GE BWRs with a range of power levels and power densities that bound the requested power uprate for VYNPS. The ODYN code has been benchmarked against BWR test data and has incorporated industry experience gained from previous transient modeling codes. ODYN uses plant specific inputs and models all the essential physical phenomena for predicting integrated plant response to the analyzed transients. Thus, the ODYN code will accurately and/or conservatively predict the integrated plant response to these transients at CPPU power levels and no new information about transient modeling is expected to be gained from performing these large transient tests.

CONCLUSION

VYNPS believes that sufficient justification has been provided to demonstrate that an MSIV closure test, turbine trip test, and generator load rejection test is not necessary or prudent. Also, the risk imposed by intentionally initiating large transient testing should not be incurred unnecessarily. As such, Entergy does not plan to perform additional large transient testing following the VYNPS CPPU.

Docket No. 50-271
BVY 03-98

Attachment

Vermont Yankee Nuclear Power Station

Technical Specification Proposed Change No. 263

Supplement No. 3

Extended Power Uprate – Updated Information

Review Guidance Matrix

Matrix 1
SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE
Materials and Chemical Engineering

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review CPPU SAR / CPPU LTR
							BWR	PWR	
Reactor Vessel Material Surveillance Program	All EPUs	EMCB	SRXB	5.3.1 Draft Rev. 2 April 1996	GDC-14 GDC-31 10 CFR 50, App. H 10 CFR 50.60	RG 1.190	2.1.1	2.1.1	3.2.1
Pressure-Temperature Limits and Upper-Shelf Energy	All EPUs	EMCB	SRXB	5.3.2 Draft Rev. 2 April 1996	GDC-14 GDC-31 10 CFR 50, App. G 10 CFR 50.60	RG 1.161 RG 1.190 RG 1.99	2.1.2	2.1.2	3.2.1
Pressurized Thermal Shock	PWR EPUs	EMCB	SRXB	5.3.2 Draft Rev. 2 April 1996	GDC-14 GDC-31 10 CFR 50.61	RG 1.190 RG 1.154		2.1.3	NA for BWRs
Reactor Internal and Core Support Materials	All EPUs	EMCB	SRXB	4.5.2 Draft Rev. 3 April 1996	GDC-1 10 CFR 50.55a	Note 1*	2.1.3	2.1.4	10.7

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review CPPU SAR / CPPU LTR
Reactor Coolant Pressure Boundary Materials	All EPUs	EMCB	EMEB SRXB	5.2.3 Draft Rev. 3 April 1996	GDC-1 10 CFR 50.55a GDC-4 GDC-14 GDC-31 10 CFR 50, App. G	RG 1.190 GL 97-01 IN 00-17s1 BL 01-01 BL 02-01 BL 02-02 Note 2* Note 3*	2.1.4	2.1.5	2.5.3, 3.2.1, 3.2.2 and 10.7
				4.5.1 Draft Rev. 3 April 1996	GDC-1 10 CFR 50.55a GDC-14				
				5.2.4 Draft Rev. 2 April 1996	10 CFR 50.55a				
				5.3.1 Draft Rev. 2 April 1996	GDC-1 10 CFR 50.55a GDC-4 GDC-14 GDC-31 10 CFR 50, App. G				
				5.3.3 Draft Rev. 2 April 1996					
				6.1.1 Draft Rev. 2 April 1996					
Leak-Before-Break	PWR EPUs	EMCB	[redacted]	3.6.3 Draft Aug. 1987	GDC-4	NUREG 1061 Vol. 3 Nov. 1984	[redacted]	2.1.6	NA for BWRs
Protective Coating Systems (Paints) - Organic Materials	All EPUs	EMCB	[redacted]	6.1.2 Draft Rev. 3 April 1996	10 CFR 50, App. B RG 1.54	[redacted]	2.1.5	2.1.7	4.2.6
Effect of EPU on Flow-Accelerated Corrosion	All EPUs	EMCB	[redacted]	[redacted]	[redacted]	Note 4*	2.1.6	2.1.8	10.7

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review CPPU SAR / CPPU LTR
Steam Generator Tube Inservice Inspection	PWR EPUs	EMCB		5.4.2.2 Draft Rev. 2 April 1996	10 CFR 50.55a	Plant TSs RG 1.121 GL 95-03 BL 88-02 GL 95-05 Note 5*		2.1.9	NA for BWRs
Steam Generator Blowdown System	PWR EPUs	EMCB		10.4.8 Draft Rev. 3 April 1996	GDC-14			2.1.10	NA for BWRs
Chemical and Volume Control System (Including Boron Recovery System)	PWR EPUs	EMCB	SPLB SRXB	9.3.4 Draft Rev. 3 April 1996	GDC-14 GDC-29			2.1.11	NA for BWRs
Reactor Water Cleanup System	BWR EPUs	EMCB		5.4.8 Draft Rev. 3 April 1996	GDC-14 GDC-60 GDC-61		2.1.7		3.11 and 10.7

Notes:

1. In addition to the SRP, guidance on neutron irradiation-related threshold for inspection for irradiation-assisted stress-corrosion cracking for BWRs is in BWRVIP-26 and for PWRs in BAW-2248 for E>1 MeV and in WCAP-14577 for E>0.1 MeV. For intergranular stress-corrosion cracking and stress-corrosion cracking in BWRs, review criteria and review guidance is contained in BWRVIP reports and associated staff safety evaluations. For thermal and neutron embrittlement of cast austenitic stainless steel, stress-corrosion cracking, and void swelling, applicants will need to provide plant-specific degradation management programs or participate in industry programs to investigate degradation effects and determine appropriate management programs.

2. For thermal aging of cast austenitic stainless steel, review guidance and criteria is contained in the May 19, 2000, letter from C. Grimes to D. Walters, "Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Components."

3. For intergranular stress corrosion cracking in BWR piping, review criteria and review guidance is contained in BWRVIP reports, NUREG-0313, Rev. 2, GL 88-01, and associated safety evaluations.

MATRIX 2

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Mechanical and Civil Engineering

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review CPPU SAR / CPPU LTR
							BWR	PWR	
Pipe Rupture Locations and Associated Dynamic Effects	All EPUs	EMEB	[redacted]	3.6.2 Draft Rev. 2 April 1996	GDC-4	[redacted]	2.2.1	2.2.1	10.1 and 10.2
Pressure-Retaining Components and Component Supports	All EPUs	EMEB	[redacted]	3.9.1 Draft Rev. 3 April 1996	GDC-1 GDC-2 GDC-14 GDC-15	[redacted]	2.2.2	2.2.2	2.5.3, 3.1, 3.2.2, 3.4, 3.5, 3.7, and 3.8
			[redacted]	3.9.2 Draft Rev. 3 April 1996	GDC-1 GDC-2 GDC-4 GDC-14 GDC-15	IN 95-016 IN 02-026			
			[redacted]	3.9.3 Draft Rev. 2 April 1996	10 CFR 50.55a GDC-1 GDC-2 GDC-4 GDC-14 GDC-15	IN 96-049 GL 96-06			
			[redacted]	5.2.1.1 Draft Rev. 3 April 1996	10 CFR 50.55a GDC-1	RG 1.84 RG 1.147 DG 1.1089 DG 1.1090 DG 1091			

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review CPPU SAR / CPPU LTR
							BWR	PWR	
Reactor Pressure Vessel Internals and Core Supports	All EPUs	EMEB	[redacted]	3.9.1 Draft Rev. 3 April 1996	GDC-1 GDC-2	[redacted]	2.2.3	2.2.3	3.1, 3.3, and 3.4.2
			[redacted]	3.9.2 Draft Rev. 3 April 1996	GDC-1 GDC-2 GDC-4	IN 95-016 IN 02-026			
			[redacted]	3.9.3 Draft Rev. 2 April 1996	10 CFR 50.55a GDC-1 GDC-2 GDC-4	IN 96-049 GL 96-06			
			[redacted]	3.9.5 Draft Rev. 3 April 1996	10 CFR 50.55a GDC-1 GDC-2 GDC-4 GDC-10	IN 02-026			
Safety-Related Valves and Pumps	All EPUs	EMEB	[redacted]	3.9.3 Draft Rev. 2 April 1996	GDC-1 10 CFR 50.55a(f)	IN 96-049 GL 96-06	2.2.4	2.2.4	3.1, 3.8, 4.1.3, 4.4.4, 4.1.6, and 4.2
			[redacted]	3.9.6 Draft Rev. 3 April 1996	GDC-1 GDC-37 GDC-40 GDC-43 GDC-46 GDC-54 10 CFR 50.55a(f)	GL 89-10 GL 95-07 GL 96-05 IN 97-090 IN 96-048s1 IN 96-048 IN 96-003 RIS 00-003 RIS 01-015 RG 1.147 RG 1.175 DG 1089 DG 1091			

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review CPPU SAR / CPPU LTR
							BWR	PWR	
Seismic and Dynamic Qualification of Mechanical and Electrical Equipment	All EPUs	EMEB	EEIB	3.10 Draft Rev. 3 April 1996	GDC-1 GDC-2 GDC-4 GDC-14 GDC-30 10 CFR 100, App. A 10 CFR 50, App. B USI A-46	[redacted]	2.2.5	2.2.5	10.1 and 10.3.3

MATRIX 3
SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE
Electrical Engineering

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review CPPU SAR / CPPU LTR
							BWR	PWR	
Environmental Qualification of Electrical Equipment	All EPUs	EEIB	[redacted]	3.11 Draft Rev. 3 April 1996	10 CFR 50.49	[redacted]	2.3.1	2.3.1	10.3.1
Offsite Power System	All EPUs	EEIB	[redacted]	8.1 Draft Rev. 3 April 1996	GDC-17	BTP PSB-1 Draft Rev. 3 April 1996 BTP ICSB-11 Draft Rev. 3 April 1996	2.3.2	2.3.2	6.1.1
				8.2 Draft Rev. 4 April 1996	GDC-17				
				8.2, App. A Draft Rev. 4 April 1996	GDC-17				
AC Onsite Power System	All EPUs	EEIB	[redacted]	8.1 Draft Rev. 3 April 1996	GDC-17	[redacted]	2.3.3	2.3.3	6.1.2
				8.3.1 Draft Rev. 3 April 1996	GDC-17				
DC Onsite Power System	All EPUs	EEIB	[redacted]	8.1 Draft Rev. 3 April 1996	GDC-17 10 CFR 50.63	[redacted]	2.3.4	2.3.4	6.2
				8.3.2 Draft Rev. 3 April 1996	GDC-17 10 CFR 50.63				

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review CPPU SAR / CPPU LTR
							BWR	PWR	
Station Blackout	All EPUs	EEIB	SPLB SRXB	8.1 Draft Rev. 3 April 1996	10 CFR 50.63	Note 1*	2.3.5	2.3.5	9.3.2
				8.2, App. B Draft Rev. 4 April 1996	10 CFR 50.63				

1. The review of station blackout includes the effects of the EPU on systems required for core cooling in the station blackout coping analysis (e.g., condensate storage tank inventory, controls and power supplies for relief valves, residual heat removing system, etc.) to ensure that the effects are accounted for in the analysis.

MATRIX 4

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Instrumentation and Controls

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review CPPU SAR / CPPU LTR
							BWR	PWR	
Reactor Trip System	All EPUs	EEIB	[redacted]	7.2 Rev. 4 June 1997	10 CFR 50.55(a)(1) 10 CFR 50.55a(h) GDC-1 GDC-4 GDC-13 GDC-19 GDC-20 GDC-21 GDC-22 GDC-23 GDC-24	[redacted]	2.4.1	2.4.1	5.3
Engineered Safety Features Systems	All EPUs	EEIB	[redacted]	7.3 Rev. 4 June 1997		[redacted]	2.4.1	2.4.1	5.3
Safety Shutdown Systems	All EPUs	EEIB	[redacted]	7.4 Rev. 4 June 1997	10 CFR 50.55(a)(1) 10 CFR 50.55a(h) GDC-1 GDC-4 GDC-13 GDC-19 GDC-24	[redacted]	2.4.1	2.4.1	5.3
Control Systems	All EPUs	EEIB	[redacted]	7.7 Rev. 4 June 1997	10 CFR 50.55(a)(1) 10 CFR 50.55a(h) GDC-1 GDC-13 GDC-19 GDC-24	[redacted]	2.4.1	2.4.1	5.1 and 5.2
Diverse I&C Systems	All EPUs	EEIB	[redacted]	7.8 Rev. 4 June 1997		[redacted]	2.4.1	2.4.1	5.3 and 9.3.1

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review CPPU SAR / CPPU LTR
							BWR	PWR	
General guidance for use of other SRP Sections related to I&C	All EPUs	EEIB		7.0 Rev. 4 June 1997					

MATRIX 5

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Plant Systems

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review CPPU SAR / CPPU LTR
							BWR	PWR	
Flood Protection	EPUs that result in significant increases in fluid volumes of tanks and vessels	SPLB		3.4.1 Rev. 2 July 1981	GDC-2		2.5.1.1.1	2.5.1.1.1	10.1.2
Equipment and Floor Drainage System	EPUs that result in increases in fluid volumes or in installation of larger capacity pumps or piping systems	SPLB		9.3.3 Rev. 2 July 1981	GDC-2 GDC-4		2.5.1.1.2	2.5.1.1.2	8.1*
Circulating Water System	EPUs that result in increases in fluid volumes associated with the circulating water system or in installation of larger capacity pumps or piping systems	SPLB		10.4.5 Rev. 2 July 1981	GDC-4		2.5.1.1.3	2.5.1.1.3	6.4.2*
Internally Generated Missiles (Outside Containment)	EPUs that result in substantially higher system pressures or changes in existing system configuration	SPLB	EMCB EMEB	3.5.1.1 Rev. 2 July 1981	GDC-4		2.5.1.2.1	2.5.1.2.1	10.1.2**
Internally Generated Missiles (Inside Containment)	EPUs that result in substantially higher system pressures or changes in existing system configuration	SPLB	EMCB EMEB	3.5.1.2 Rev. 2 July 1981	GDC-4		2.5.1.2.1	2.5.1.2.1	10.1.2**

*NO IGNIFICANT INCREASE IN FLUID VOLUMES

**NO INCREASE IN SYSTEM PRESSURE

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review CPPU SAR / CPPU LTR
							BWR	PWR	
Protection Against Postulated Piping Failures in Fluid Systems Outside Containment	EPUs that affect environmental conditions, habitability of the control room, or access to areas important to safe control of postaccident operations	SPLB	EMCB EMEB	3.6.1 Rev. 1 July 1981	GDC-4		2.5.1.3	2.5.1.3	10.1 and 10.2
Fire Protection Program	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		9.5.1 Rev. 3 July 1981	10 CFR 50.48 10 CFR 50, App. R GDC-3 GDC-5	Note 1*	2.5.1.4	2.5.1.4	6.7
PWR Dry Containments, Including Subatmospheric Containments	EPUs for PWR plants with dry containments (including subatmospheric containments) except where the application demonstrates that previous analysis is bounding	SPLB		6.2.1 Rev. 2 July 1981 6.2.1.1.A Rev. 2 July 1981	GDC-13 GDC-16 GDC-38 GDC-50 GDC-64			2.5.2.1	NA for BWRs
Ice Condenser Containments	EPUs for PWR plants with ice condenser containments except where the application demonstrates that previous analysis is bounding	SPLB		6.2.1 Rev. 2 July 1981 6.2.1.1.B Rev. 2 July 1981	GDC-13 GDC-16 GDC-38 GDC-50 GDC-64			2.5.2.1	NA for BWRs
Pressure-Suppression Type BWR Containments	EPUs for BWR plants with pressure-suppression containments except where the application demonstrates that previous analysis is bounding	SPLB		6.2.1 Rev. 2 July 1981 6.2.1.1.C Rev. 6 Aug. 1984	GDC-4 GDC-13 GDC-16 GDC-50 GDC-64		2.5.2.1		4.1 through 4.1.2

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review CPPU SAR / CPPU LTR
Subcompartment Analysis	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		6.2.1 Rev. 2 July 1981 6.2.1.2 Rev. 2 July 1981	GDC-4 GDC-50		2.5.2.2	2.5.2.2	4.1.2.3
Mass and Energy Release Analysis for Postulated Loss-of-Coolant	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		6.2.1 Rev. 2 July 1981 6.2.1.3 Rev. 1 July 1981	GDC-50 10 CFR 50, App. K		2.5.2.3.1	2.5.2.3.1	4.1.1 through 4.1.2.2
Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures	PWR EPUs except where the application demonstrates that previous analysis is bounding	SPLB		6.2.1 Rev. 2 July 1981 6.2.1.4 Rev. 1 July 1981	GDC-50			2.5.2.3.2	NA for BWRs
Combustible Gas Control In Containment	EPUs that impact hydrogen release assumptions	SPLB		6.2.5 Rev. 2 July 1981	10 CFR 50.44 10 CFR 50.46 GDC-5 GDC-41 GDC-42 GDC-43		2.5.2.4	2.5.2.4	4.7
Containment Heat Removal	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		6.2.2 Rev. 4 Oct. 1985	GDC-38	DG-1107	2.5.2.5	2.5.2.5	3.10
Secondary Containment Functional Design	EPUs that affect the pressure and temperature response, or draw-down time of the secondary containment	SPLB		6.2.3 Rev. 2 July 1981	GDC-4 GDC-16		2.5.2.6		4.5

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review CPPU SAR / CPPU LTR
Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies	PWR EPUs except where the application demonstrates that previous analysis is bounding	SPLB	SRXB	6.2.1 Rev. 2 July 1981 6.2.1.5 Rev. 2 July 1981	10 CFR 50.46 10 CFR 50, App. K	[redacted]	[redacted]	2.5.2.6	NA for BWRs
Pressurizer Relief Tank	PWR EPUs that affect pressurizer discharge to the PRT	SPLB	EMEB	5.4.11 Rev. 2 July 1981	GDC-2 GDC-4	[redacted]	[redacted]	2.5.2.7	NA for BWRs
Control Room Habitability System	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB	SPSB	6.4 Draft Rev. 3 April 1996	GDC-4 GDC-19	Note 2* Note 3*	2.5.3.1	2.5.3.1	4.4
ESF Atmosphere Cleanup System	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB	SPSB	6.5.1 Rev. 2 July 1981	GDC-19 GDC-41 GDC-61 GDC-64	[redacted]	2.5.3.2	2.5.3.2	4.5
Fission Product Control Systems and Structures	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB	EMCB	6.5.3 Rev. 2 July 1981	GDC-41	[redacted]	2.5.3.3	2.5.3.3	4.5
Main Condenser Evacuation System	EPUs for which the main condenser evacuation system is modified	SPLB	[redacted]	10.4.2 Rev. 2 July 1981	GDC-60 GDC-64	[redacted]	2.5.3.4	2.5.3.4	7.2 (no mod)
Turbine Gland Sealing System	EPUs for which the turbine gland sealing system is modified	SPLB	[redacted]	10.4.3 Rev. 2 July 1981	GDC-60 GDC-64	[redacted]	2.5.3.5	2.5.3.5	7.1 (no mod)
Main Steam Isolation Valve Leakage Control System	BWR EPU that affect the amount of valve leakage that is assumed and resultant dose consequences.	SPLB	[redacted]	6.7 Rev. 2 July 1981	GDC-54	[redacted]	2.5.3.6	[redacted]	4.6
Control Room Area Ventilation System	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB	SPSB	9.4.1 Rev. 2 July 1981	GDC-4 GDC-19 GDC-60	[redacted]	2.5.4.1	2.5.4.1	4.4

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review CPPU SAR / CPPU LTR
Spent Fuel Pool Area Ventilation System	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB	SPSB	9.4.2 Rev. 2 July 1981	GDC-60 GDC-61		2.5.4.2	2.5.4.2	6.6
Auxiliary and Radwaste Area Ventilation System	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		9.4.3 Rev. 2 July 1981	GDC-60		2.5.4.3	2.5.4.3	6.6
Turbine Area Ventilation System	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		9.4.4 Rev. 2 July 1981	GDC-60		2.5.4.3	2.5.4.3	6.6
ESF Ventilation System	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		9.4.5 Rev. 2 July 1981	GDC-4 GDC-17 GDC-60		2.5.4.4	2.5.4.4	6.6
Spent Fuel Pool Cooling and Cleanup System	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB	EMCB	9.1.3 Rev. 1 July 1981	GDC-5 GDC-44 GDC-61	Note 4*	2.5.5.1	2.5.5.1	6.3
Station Service Water System	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		9.2.1 Rev. 4 June 1985	GDC-4 GDC-5 GDC-44	GL 89-13 and Suppl. 1 GL 96-06 and Suppl. 1	2.5.5.2	2.5.5.2	6.4.1 and 6.4.4
Reactor Auxiliary Cooling Water Systems	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		9.2.2 Rev. 3 June 1986	GDC-4 GDC-5 GDC-44	GL 89-13 and Suppl. 1 GL 96-06 and Suppl. 1	2.5.5.3	2.5.5.3	6.4.3
Ultimate Heat Sink	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		9.2.5 Rev. 2 July 1981	GDC-5 GDC-44		2.5.5.4	2.5.5.4	6.4.5

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review CPPU SAR / CPPU LTR
Auxiliary Feedwater System	PWR EPUs except where the application demonstrates that previous analysis is bounding	SPLB		10.4.9 Rev. 2 July 1981	GDC-4 GDC-5 GDC-19 GDC-34 GDC-44			2.5.5.5	NA for BWRs
Main Steam Supply System	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		10.3 Rev. 3 April 1984	GDC-4 GDC-5 GDC-34		2.5.6.1	2.5.6.1	3.5.2 and 7.3
Main Condenser	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		10.4.1 Rev. 2 July 1981	GDC-60		2.5.6.2	2.5.6.2	7.2
Turbine Bypass System	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		10.4.4 Rev. 2 July 1981	GDC-4 GDC-34		2.5.6.3	2.5.6.3	7.3
Condensate and Feedwater System	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		10.4.7 Rev. 3 April 1984	GDC-4 GDC-5 GDC-44		2.5.6.4	2.5.6.4	7.4
Gaseous Waste Management Systems	EPUs that impact the level of fission products in the reactor coolant system, or the amount of gaseous waste	SPLB	IEHB	11.3 Draft Rev. 3 April 1996	10 CFR 20.1302 GDC-3 GDC-60 GDC-61 10 CFR 50, App. I		2.5.7.1	2.5.7.1	8.2
Liquid Waste Management Systems	EPUs that impact the level of fission products in the reactor coolant system, or the amount of liquid waste	SPLB	IEHB	11.2 Draft Rev. 3 April 1996	10 CFR 20.1302 GDC-60 GDC-61 10 CFR 50, App. I		2.5.7.2	2.5.7.2	8.1
Solid Waste Management Systems	EPUs that impact the level of fission products in the reactor coolant system, or the amount of solid waste	SPLB	IEHB	11.4 Draft Rev. 3 April 1996	10 CFR 20.1302 GDC-60 GDC-63 GDC-64 10 CFR 71		2.5.7.3	2.5.7.3	8.1

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review CPPU SAR / CPPU LTR
Emergency Diesel Engine Fuel Oil Storage and Transfer System	EPUs that result in higher EDG electrical demands	SPLB	[redacted]	9.5.4 Rev. 2 July 1981	GDC-4 GDC-5 GDC-17	[redacted]	2.5.8.1	2.5.8.1	6.1.1
Light Load Handling System (Related to Refueling)	EPUs except where the application demonstrates that previous analysis is bounding	SPLB	SPSB	9.1.4 Rev. 2 July 1981	GDC-61 GDC-62	[redacted]	2.5.8.2	2.5.8.2	6.8

Notes:
1. Supplemental guidance for review of fire protection is provided in Attachment 2 to this matrix.

MATRIX 6

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Reactor Systems

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review CPPU SAR / CPPU LTR
							BWR	PWR	
Fuel System Design	All EPUs	SRXB	[redacted]	4.2 Draft Rev. 3 April 1996	10 CFR 50.46 GDC-10 GDC-27 GDC-35	Note 1* Note 2*	2.6.1	2.6.1	2.1
Nuclear Design	All EPUs	SRXB	[redacted]	4.3 Draft Rev. 3 April 1996	GDC-10 GDC-11 GDC-12 GDC-13 GDC-20 GDC-25 GDC-26 GDC-27 GDC-28	RG 1.190 GSI 170 IN 97-085	2.6.2	2.6.2	2.2, 2.3, and 2.4
Thermal and Hydraulic Design	All EPUs	SRXB	[redacted]	4.4 Draft Rev. 2 April 1996	GDC-10 GDC-12	Note 3*	2.6.3	2.6.3	2.2, 2.3, and 2.4
Functional Design of Control Rod Drive System	All EPUs	SRXB	SPLB	4.6 Draft Rev. 2 April 1996	GDC-4 GDC-23 GDC-25 GDC-26 GDC-27 GDC-28 GDC-29 10 CFR 50.62(c)(3)	[redacted]	2.6.4.1	2.6.4.1	2.5

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review CPPU SAR / CPPU LTR
							BWR	PWR	
Overpressure Protection during Power Operation	All EPUs	SRXB		5.2.2 Draft Rev. 3 April 1996	GDC-15 GDC-31	Note 4*	2.6.4.2	2.6.4.2	3.1
Overpressure Protection during Low Temperature Operation	PWR EPUs	SRXB		5.2.2 Draft Rev. 3 April 1996	GDC-15 GDC-31			2.6.4.3	NA for BWRs
Reactor Core Isolation Cooling System	BWR EPUs	SRXB		5.4.6 Draft Rev. 4 April 1996	GDC-4 GDC-5 GDC-29 GDC-33 GDC-34 GDC-54 10 CFR 50.63		2.6.4.3		3.9
Residual Heat Removal System	All EPUs	SRXB		5.4.7 Draft Rev. 4 April 1996	GDC-4 GDC-5 GDC-19 GDC-34	Note 5*	2.6.4.4	2.6.4.4	3.10
Emergency Core Cooling System	All EPUs	SRXB		6.3 Draft Rev. 3 April 1996	GDC-4 GDC-27 GDC-35 10 CFR 50.46 10 CFR 50 App. K	Note 6*	2.6.5.6.2	2.6.5.6.3	4.2 and 4.3
Standby Liquid Control System	BWR EPUs	SRXB	EMCB SPLB	9.3.5 Draft Rev. 3 April 1996	GDC-26 GDC-27 10 CFR 50.62(c)(4)	Note 12*	2.6.4.5		6.5
Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve	All EPUs	SRXB		15.1.1-4 Draft Rev. 2 April 1996	GDC-10 GDC-15 GDC-20 GDC-26	Note 7*	2.6.5.1	2.6.5.1.1	9.1

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review CPPU SAR / CPPU LTR
							BWR	PWR	
Steam System Piping Failures Inside and Outside of Containment	PWR EPUs	SRXB		15.1.5 Draft Rev. 3 April 1996	GDC-27 GDC-28 GDC-31 GDC-35	Note 7*		2.6.5.1.2	NA for BWRs
Loss of External Load; Turbine Trip, Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve (BWR); and Steam Pressure Regulator Failure (Closed)	All EPUs	SRXB		15.2.1-5 Draft Rev. 2 April 1996	GDC-10 GDC-15 GDC-26	Note 7*	2.6.5.2.1	2.6.5.2.1	3.1 and 9.1
Loss of Nonemergency AC Power to the Station Auxiliaries	All EPUs	SRXB		15.2.6 Draft Rev. 2 April 1996	GDC-10 GDC-15 GDC-26	Note 7*	2.6.5.2.2	2.6.5.2.2	3.1 and 9.1
Loss of Normal Feedwater Flow	All EPUs	SRXB	EEIB	15.2.7 Draft Rev. 2 April 1996	GDC-10 GDC-15 GDC-26	Note 7*	2.6.5.2.3	2.6.5.2.3	9.1
Feedwater System Pipe Breaks Inside and Outside Containment	PWR EPUs	SRXB	EEIB	15.2.8 Draft Rev. 2 April 1996	GDC-27 GDC-28 GDC-31 GDC-35	Note 7*		2.6.5.2.4	NA for BWRs
Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions	All EPUs	SRXB		15.3.1-2 Draft Rev. 2 April 1996	GDC-10 GDC-15 GDC-26	Note 7*	2.6.5.3.1	2.6.5.3.1	9.1
Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break	All EPUs	SRXB		15.3.3-4 Draft Rev. 3 April 1996	GDC-27 GDC-28 GDC-31	Note 7*	2.6.5.3.2	2.6.5.3.2	9.1
Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition	All EPUs	SRXB		15.4.1 Draft Rev. 3 April 1996	GDC-10 GDC-20 GDC-25	Note 7*	2.6.5.4.1	2.6.5.4.1	5.3.4 and 9.1

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review CPPU SAR / CPPU LTR
							BWR	PWR	
Uncontrolled Control Rod Assembly Withdrawal at Power	All EPUs	SRXB	[redacted]	15.4.2 Draft Rev. 3 April 1996	GDC-10 GDC-20 GDC-25	Note 7*	2.6.5.4.2	2.6.5.4.2	5.3.5 and 9.1
Control Rod Misoperation (System Malfunction or Operator Error)	PWR EPUs	SRXB	[redacted]	15.4.3 Draft Rev. 3 April 1996	GDC-10 GDC-20 GDC-25	Note 7*	[redacted]	2.6.5.4.3	NA for BWRs
Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate	All EPUs	SRXB	[redacted]	15.4.4-5 Draft Rev. 2 April 1996	GDC-10 GDC-15 GDC-20 GDC-26 GDC-28	Note 7*	2.6.5.4.3	2.6.5.4.4	9.1
Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant	PWR EPUs	SRXB	[redacted]	15.4.6 Draft Rev. 2 April 1996	GDC-10 GDC-15 GDC-26	Note 7*	[redacted]	2.6.5.4.5	NA for BWRs
Spectrum of Rod Ejection Accidents	PWR EPUs	SRXB	[redacted]	15.4.8 Draft Rev. 3 April 1996	GDC-28	Note 7*	[redacted]	2.6.5.4.6	NA for BWRs
Spectrum of Rod Drop Accidents	BWR EPUs	SRXB	[redacted]	15.4.9 Draft Rev. 3 April 1996	GDC-28	Note 7*	2.6.5.4.4	[redacted]	9.2
Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory	All EPUs	SRXB	[redacted]	15.5.1-2 Draft Rev. 2 April 1996	GDC-10 GDC-15 GDC-26	Note 7* Note 8*	2.6.5.5	2.6.5.5	9.1
Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve or a BWR Pressure Relief Valve	All EPUs	SRXB	[redacted]	15.6.1 Draft Rev. 2 April 1996	GDC-10 GDC-15 GDC-26	Note 7*	2.6.5.6.1	2.6.5.6.1	9.1

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review CPPU SAR / CPPU LTR
Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve or a BWR Pressure Relief Valve	All EPUs	SRXB		15.6.1 Draft Rev. 2 April 1996	GDC-10 GDC-15 GDC-26	Note 7*	2.6.5.6.1	2.6.5.6.1	9.1
Steam Generator Tube Rupture	PWR EPUs	SRXB		15.6.3 Draft Rev. 3 April 1996	Note 7* Note 9*	Note 7* Note 9*		2.6.5.6.2	NA for BWRs
Loss-of Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary	All EPUs	SRXB		15.6.5 Draft Rev. 3 April 1996	GDC-35 10 CFR 50.46	Note 7* Note 10*	2.6.5.6.2	2.6.5.6.3	4.3 and 9.2
Anticipated Transient Without Scram	All EPUs	SRXB				Note 7* Note 11* Note 12*	2.6.5.7	2.6.5.7	9.3
New Fuel Storage	EPU applications that request approval for new fuel.	SRXB		9.1.1 Draft Rev. 3 April 1996	GDC-62		2.6.6.1	2.6.6.1	1.2.3 and 2.1*
Spent Fuel Storage	EPU applications that request approval for new fuel.	SRXB		9.1.2 Draft Rev. 4 April 1996	GDC-4 GDC-62		2.6.6.2	2.6.6.2	1.2.3 and 2.1*

Notes:

1. When mixed cores (i.e., fuels of different designs) are used, the review covers the licensee's evaluation of the effects of mixed cores on design-basis accident and transient analyses.

2. The current acceptance criteria for fuel damage for reactivity insertion accidents (RIAs) requires revision per Research Information Letter No. 174, "Interim Assessment of Criteria for Analyzing Reactivity Accidents at High Burnup." The Office of Nuclear Regulatory Research is conducting confirmatory research on RIAs and the Office of Nuclear Reactor Regulation is discussing the issue of fuel damage criteria with the nuclear power industry as part of the industry's proposal to increase fuel burnup limits in the future. In the interim, current methods for assessing fuel damage in RIAs are considered acceptable based on the NRC staff's understanding of actual fuel performance, as shown in three-dimensional kinetic calculations which indicate acceptably low fuel cladding enthalpy.

MATRIX 7

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Source Terms and Radiological Consequences Analyses

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review CPPU SAR / CPPU LTR
							BWR	PWR	
Source Terms for Input into Radwaste Management Systems Analyses	All EPUs	SPSB		11.1 Draft Rev. 3 April 1996	10 CFR 20 10 CFR 50, App. I GDC-60		2.7.1	2.7.1	8.4
Radiological Consequence Analyses Using Alternative Source Terms	EPUs that utilize alternative source term	SPSB	EEIB EMCB EMEB IEHB SPLB SRXB	15.0.1 Rev. 0 July 2000	10 CFR 50.67 GDC-19 10 CFR 50.49 10 CFR 51 10 CFR 50, App. E NUREG-0737		2.7.2	2.7.2	9.2
Radiological Consequences of Main Steamline Failures Outside Containment for a PWR	PWR EPUs that do not utilize alternative source term whose main steamline break analyses result in fuel failure	SPSB	SRXB	15.1.5, App. A Draft Rev. 3 April 1996	10 CFR 100	Notes 4, 5, 6, 7, 27*		2.7.2	NA for BWRs
				6.4 Draft Rev. 3 April 1996	GDC-19	Notes 1, 2, 3, 28, 29*			
Radiological Consequences of Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break	EPUs that do not utilize alternative source term whose reactor coolant pump rotor seizure or reactor coolant pump shaft break results in fuel failure	SPSB	SRXB	15.3.3-4 Draft Rev. 3 April 1996	10 CFR 100	Notes 5, 8, 9, 27*		2.7.3	NA for BWRs
				6.4 Draft Rev. 3 April 1996	GDC-19	Notes 1, 2, 3, 28, 29*			

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review CPPU SAR / CPPU LTR
							BWR	PWR	
Radiological Consequences of a Control Rod Ejection Accident	PWR EPUs that do not utilize alternative source term whose rod ejection accident results in fuel failure or melting	SPSB	SRXB	15.4.8, App. A Draft Rev. 2 April 1996	10 CFR 100	Notes 4, 21, 22, 27*		2.7.4	NA for BWRs
				6.4 Draft Rev. 3 April 1996	GDC-19	Notes 1, 2, 3, 28, 29*			
Radiological Consequences of Control Rod Drop Accident	BWR EPUs that do not utilize alternative source term whose control rod drop accident results in fuel failure or melting	SPSB	SRXB	15.4.9, App. A Draft Rev. 3 April 1996	10 CFR 100	Notes 9, 10, 27*	2.7.2		9.2
				6.4 Draft Rev. 3 April 1996	GDC-19	Notes 1, 2, 3, 28, 29*			
Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment	EPUs that do not utilize alternative source term whose failure of small lines carrying primary coolant outside containment result in fuel failure	SPSB		15.6.2 Draft Rev. 3 April 1996	GDC-55 10 CFR 100		2.7.3	2.7.5	9.2
				6.4 Draft Rev. 3 April 1996	GDC-19	Notes 1, 2, 3, 28, 29*			
Radiological Consequences of Steam Generator Tube Failure	PWR EPUs that do not utilize alternative source term whose steam generator tube failure results in fuel failure	SPSB	SRXB	15.6.3 Draft Rev. 3 April 1996	10 CFR 100	Notes 4, 13, 14, 15, 27*		2.7.6	NA for BWRs
				6.4 Draft Rev. 3 April 1996	GDC-19	Notes 1, 2, 3, 28, 29*			
Radiological Consequences of Main Steamline Failure Outside Containment for a BWR	BWR EPUs that do not utilize alternative source term whose main steam line failure outside containment results in fuel failure	SPSB	SRXB	15.6.4 Draft Rev. 3 April 1996	10 CFR 100	Note 27*	2.7.4		9.2
				6.4 Draft Rev. 3 April 1996	GDC-19	Notes 1, 2, 3, 28, 29*			

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review CPPU SAR / CPPU LTR
							BWR	PWR	
Radiological Consequences of a Design Basis Loss-Of-Coolant-Accident Including Containment Leakage Contribution	EPUs that do not utilize alternative source term	SPSB	SPLB	15.6.5, App. A Draft Rev. 2 April 1996	10 CFR 100	Notes 4, 23, 24, 25, 26, 27*	2.7.5	2.7.7	9.2
				6.4 Draft Rev. 3 April 1996	GDC-19	Notes 1, 2, 3, 28, 29*			
Radiological Consequences of a Design Basis Loss-Of-Coolant-Accident: Leakage from ESF Components Outside Containment	EPUs that do not utilize alternative source term	SPSB	SPLB	15.6.5, App. B Draft Rev. 2 April 1996	10 CFR 100	Notes 11, 27*	2.7.5	2.7.7	9.2
				6.4 Draft Rev. 3 April 1996	GDC-19	Notes 1, 2, 3, 28, 29*			
Radiological Consequences of a Design Basis Loss-Of-Coolant-Accident: Leakage from Main Steam Isolation Valves	BWR EPUs that do not utilize alternative source term	SPSB		15.6.5, App. D Draft Rev. 2 April 1996	10 CFR 100	Notes 9, 12, 27*	2.7.5		9.2
				6.4 Draft Rev. 3 April 1996	GDC-19	Notes 1, 2, 3, 28, 29*			
Radiological Consequences of Fuel Handling Accidents	EPUs that do not utilize alternative source term	SPSB	SPLB	15.7.4 Draft Rev. 2 April 1996	10 CFR 100 GDC-61	Notes 4, 5, 18, 19, 20, 27*	2.7.6	2.7.8	9.2
				6.4 Draft Rev. 3 April 1996	GDC-19	Notes 1, 2, 3, 28, 29*			
Radiological Consequences of Spent Fuel Cask Drop Accidents	EPUs that do not utilize alternative source term	SPSB	EMEB SPLB	15.7.5 Draft Rev. 3 April 1996	10 CFR 100 GDC-61	Notes, 5, 16, 17, 8, 18, 27*	2.7.7	2.7.9	NA (VYNPS utilizes AST)
				6.4 Draft Rev. 3 April 1996	GDC-19	Notes 1, 2, 3, 28, 29*			

MATRIX 8

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Health Physics

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review CPPU SAR / CPPU LTR
							BWR	PWR	
Radiation Sources	All EPUs	IEHB	[redacted]	12.2 Draft Rev. 3 April 1996	10 CFR 20	[redacted]	2.8.1	2.8.1	8.3 and 8.4
Radiation Protection Design Features	All EPUs	IEHB	[redacted]	12.3-4 Draft Rev. 3 April 1996	10 CFR 20 GDC-19	[redacted]	2.8.1	2.8.1	8.5
Operational Radiation Protection Program	All EPUs	IEHB	[redacted]	12.5 Draft Rev. 3 April 1996	10 CFR 20	Note 1*	2.8.1	2.8.1	8.5

Notes:

1. Regulatory Guide 8.14 was withdrawn on February 9, 2001, and should not be used.

MATRIX 9

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Human Performance

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review CPPU SAR / CPPU LTR
							BWR	PWR	
Reactor Operator Training	All EPUs	IEHB		13.2.1 Draft Rev. 2 Dec. 2002	Specific review questions are provided in the template safety evaluations.		2.9	2.9	10.6
Training for Non-Licensed Plant Staff	All EPUs	IEHB		13.2.2 Draft Rev. 2 Dec. 2002	Specific review questions are provided in the template safety evaluations.		2.9	2.9	10.6
Operating and Emergency Operating Procedures	All EPUs	IEHB	SPLB SRXB	13.5.2.1 Draft Rev. 1 Dec. 2002	Specific review questions are provided in the template safety evaluations.		2.9	2.9	10.9
Human Factors Engineering	All EPUs	IEHB		18.0 Draft Rev. 1 Dec. 2002	Specific review questions are provided in the template safety evaluations.		2.9	2.9	10.6

MATRIX 10

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Power Ascension and Testing Plan

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review CPPU SAR / CPPU LTR
							BWR	PWR	
Power Ascension and Testing	All EPUs	IEHB	EEIB EMCB EMEB SPLB SPSB SRXB	14.2.1 Draft Rev. 0 Dec. 2002	Entire Section	[redacted]	2.10	2.10	10.4

MATRIX 11

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Risk Evaluation

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review CPPU SAR / CPPU LTR
							BWR	PWR	
Risk Evaluation	All EPUs	SPSB	[redacted]	[redacted]	[redacted]	Note 1* RG 1.174 RIS 2001-02	2.11	2.11	10.5

Notes:
1. The staff's review is based on Attachment 2 to this matrix. Attachment 2 invokes SRP Chapter 19, Appendix D, if special circumstances are identified during the review.