Entergy

Entergy Nuclear Operations, Inc.
Vermont Yankee
P.O. Box 0500
185 Old Ferry Road
Brattleboro, VT 05302-0500
Tel 802 257 5271

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BVY 06-080
TAC No. MC0761

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Reference: (1) U.S. Nuclear Regulatory Commission (Richard B. Ennis) letter to Entergy (Michael Kansler), "Vermont Yankee Nuclear Power Station – Issuance of Amendment Re: Extended Power Uprate (TAC No. MC0761)," NVY 06-028, dated March 2, 2006.

Subject: **Vermont Yankee Nuclear Power Station**
License No. DPR-28 (Docket No. 50-271)
Extended Power Uprate – License Amendment No. 229
Clarifications to NRC's Final Safety Evaluation

On March 2, 2006, the NRC issued Amendment No. 229 to the operating license for the Vermont Yankee Nuclear Power Station (VYNPS) authorizing an increase in the maximum authorized power level of the VYNPS from 1593 megawatts thermal (MWt) to 1912 MWt (Reference 1). Entergy's review of NRC's final safety evaluation (FSE) associated with issuance of the license amendment identified a number of items that should be clarified in regard to the licensing basis for VYNPS and extended power uprate operation.

Attachment 1 to this letter provides clarifications to the FSE. A markup of the affected pages of the FSE to assist in identification of the applicable statements is provided in Attachment 2. The clarifications do not invalidate the conclusions documented in the FSE.

There are no new regulatory commitments contained in this submittal.

If you have any questions, please contact Mr. James M. DeVincentis at (802) 258-4236.

Sincerely,

for

Ted A. Sullivan
Site Vice President
Vermont Yankee Nuclear Power Station

Attachments (2)
cc listing (next page)

A001

cc: Mr. Samuel J. Collins (w/o attachments)
Regional Administrator, Region 1
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406-1415

Mr. Richard B. Ennis, Project Manager
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Mail Stop O-8-B1
Washington, DC 20555

USNRC Resident Inspector (w/o attachments)
Entergy Nuclear Vermont Yankee, LLC
P.O. Box 157
Vernon, Vermont 05354

Mr. David O'Brien, Commissioner
VT Department of Public Service
112 State Street – Drawer 20
Montpelier, Vermont 05620-2601

BVY 06-080
Docket No. 50-271

Attachment 1

Vermont Yankee Nuclear Power Station

Extended Power Uprate

Clarifications to NRC's Final Safety Evaluation

Total number of pages in Attachment 1 (excluding this cover sheet) is 5.

CLARIFICATIONS TO NRC'S FINAL SAFETY EVALUATION FOR EXTENDED POWER UPRATE

	Location[1]	Existing Text	Recommended Text	Basis
1	Section 1.4 Plant Modifications Page 6	Added a second primary protection scheme on the VYNPS main **generator.**	Added a second primary protection scheme on the VYNPS main **transformer.**	Plant minor modification. ISO-New England System Impact Study (p. viii).
2	Section 2.3.2.2 Main Generator Page 56	No generator protective relay changes are necessary, however, some protective relay setpoints will be modified for the rewound generator rating.	A generator out-of-step relay was added to protect both the main generator and the transmission system from potential damaging out-of-step conditions. Additionally, the bushing current transformers (CTs) have been replaced for the EPU, and some protective relay setpoints were modified for the rewound generator rating.	Plant minor modification. Make consistent with FSE section 1.4 and ISO-New England System Impact Study.
3	Section 2.3.2.6 Startup Transformers Page 57	...the loading on one transformer is **17.9** MVA and the loading on the other is **24.8** MVA.	...the loading on one transformer is **17.6** MVA and the loading on the other is **26.0** MVA.	Calculation VYC-1088 R3, MCC 04
4	Section 2.4.1 Reactor Protection, Safety Features Actuation and Control Systems Page 72	Recirculation Pump Net Positive Suction Head (NPSH) Trip	(delete)	There is no recirculation pump NPSH trip feature.
5	Section 2.4.1 Reactor Protection, Safety Features Actuation and Control Systems Page 73	Add a second pressure switch to each pump to provide signal for *recirculation runback* on loss of condensate pump.	Add a second pressure switch to each pump to provide signal for reactor feedwater pump trip on loss of condensate pump.	Specify direct action of pressure switches (pump trip). Pump breaker trip initiates recirculation runback.

[1] Page numbers correspond to NRC's Final Safety Evaluation for VYNPS extended power uprate as published in ADAMS (ascension number ML060050028)a

CLARIFICATIONS TO NRC'S FINAL SAFETY EVALUATION FOR EXTENDED POWER UPRATE

	Location[1]	Existing Text	Recommended Text	Basis
6	Section 2.4.1 Reactor Protection, Safety Features Actuation and Control Systems Page 73	Condensate **Heater** Pressure Low	Condensate **Header** Pressure Low	Typographical error.
7	Section 2.5.3.2 Service Water Page 83	Except for the SBO and Appendix R events, the licensee's analyses for EPU operation use the same SWS flow rates that are credited for the current licensed power level...	Except for the SBO, Appendix R, and **stuck open relief valve** events, the licensee's analyses for EPU operation use the same SWS flow rates that are credited for the current licensed power level...	Analysis of the stuck open relief valve event also credits two RHRSW pumps.
8	Section 2.8.1 Fuel System Design Page 147	The peak bundle power will increase from 7.02 MWt before the EPU to 7.37 MWt after the EPU.	The peak bundle power may increase slightly for EPU.	The 7.02 MWt for original licensed thermal power and the 7.37 MWt for EPU are representative values—actual values may be slightly higher or lower. The 7.37 MWt peak bundle power for EPU was not meant to be a limit.
9	Section 2.8.1 Fuel System Design Page 148	In general, the licensee must ensure that plant operation is in compliance with the cycle-specific thermal limits **(SLMCPR, OLMCPR, MAPLHGR, and maximum LHGR)** and specify the thermal limits in a cycle-specific COLR as required by VYNPS TSs.	In general, the licensee must ensure that plant operation is in compliance with the cycle-specific thermal limits **(MCPR, APLHGR, and LHGR)** and specify the thermal limits in a cycle-specific COLR as required by VYNPS TSs.	SLMCPR is not required to be in the COLR. Thermal limit designations made consistent with the requirements of TS Section 6.6.C.

CLARIFICATIONS TO NRC'S FINAL SAFETY EVALUATION FOR EXTENDED POWER UPRATE

	Location[1]	Existing Text	Recommended Text	Basis
10	Section 2.8.4.1 Functional Design of Control Rod Drive System Pages 153 and 156	(1) draft GDC-40 and 42, insofar as they require that protection for ESFs against the dynamic effects and missiles that might result from plant equipment failures, as well as the effects of a LOCA; ...the CRD system and associated analyses will continue to meet the requirements of draft GDC...40, and 42...	(delete) and re-number remaining items. (delete GDC 40 and 42)	The control rod drive system is not an ESF; therefore, GDC-40 and 42 do not apply.
11	Section 2.8.4.3 Reactor Core Isolation Cooling System Pages 158 and 160	(1) draft GDC-40 and 42, insofar as they require that protection for ESFs against the dynamic effects and missiles that might result from plant equipment failures, as well as the effects of a LOCA; ...the RCIC system and associated analyses will continue to meet the requirements of draft GDC...40, and 42...	(delete) and re-number remaining items. (delete GDC 40 and 42)	The Reactor Core Isolation Cooling system is not an ESF; therefore, GDC-40 and 42 do not apply.
12	Section 2.8.4.4 Residual Heat Removal System Page 161	The peak suppression pool temperature during a limiting LOCA remains below the RHR pump seal design temperature of 210°F.	The peak suppression pool temperature during a limiting LOCA remains below the RHR pump seal design temperature of 197°F.	RHR pump seal design temperature is 197°F. The Core Spray pump seal design temperature is 210°F.
13	Section 2.8.7.2.2.3.2 LHGR Limit Page 197	(The table of LHGR limits for GE-14 fuel contains an erroneous value. Because it is proprietary information of GE, that value is not repeated here.)	(Change value to match value in Table 2 of GE report DB-0012.03 R0, on sheet 17 of 22.)	The subject value in the FSE is not a limit, but may be a design value.

CLARIFICATIONS TO NRC'S FINAL SAFETY EVALUATION FOR EXTENDED POWER UPRATE

	Location[1]	Existing Text	Recommended Text	Basis
14	Section 2.8.7.2.2.3.6 Void Coefficient Page 208	A plot showing the potential errors is presented in Figures 2.8.7-7 through 2.8.7-11.	A plot showing the potential errors is presented in Figures 2.8.7-12 through 2.8.7-16.	Typographical error.
15	Section 2.8.7.2.2.3.7 Code-to-Code Comparisons Page 210	Figure 2.8.7-23 shows the root-mean-square (RMS) of the difference between TGBLA-6 and CASMO-4. The RMS error is used in development of the SLMCPR.	Figure 2.8.7-23 shows the root-mean-square (RMS) of the difference between TGBLA-6 and CASMO-4. A similar RMS error (between TGBLA-6 and MCNP) is used in development of the SLMCPR.	The RMS error shown in this figure is NOT used in the development of SLMCPR. The RMS difference between TGBLA-6 and CASMO-4 is simply used to support overall conclusions.
16	Page 253	The licensee has estimated that almost 90% of this dose, 13.4 mrem per year, is due to N-16 skyshine from turbine building components.	Change 90% to 80%.	Measurements along fence line have shown approximately a 20/80 (direct vs. skyshine) relationship.
17	Section 2.11.1 Human Factors Page 255	Because no new procedures would be required...	(delete)	New procedures were developed to support EPU operation.
18	Section 2.11.1 Human Factors Page 256	...provide for automatic recirculation runback given a single reactor feedwater pump (RFP) trip under EPU conditions, and this enhancement can be regarded essentially as the automation of an operator action.	...provide for automatic recirculation runback given a single reactor feedwater pump (RFP) **or condensate pump** trip under EPU conditions, and this enhancement can be regarded essentially as the automation of an operator action.	A trip of either a RFP or condensate pump will result in an automatic recirculation runback at EPU conditions.
19	Section 2.11.1 Human Factors Page 258	...and the condensate flow recorder would be re-scaled.	(delete)	There is no condensate flow recorder.

CLARIFICATIONS TO NRC'S FINAL SAFETY EVALUATION FOR EXTENDED POWER UPRATE

	Location[1]	Existing Text	Recommended Text	Basis
20	Section 2.13.1 Risk Evaluation of Extended Power Uprate Section Page 276	Setpoint Changes Turbine overspeed	(Delete "turbine overspeed" setpoint change.)	The turbine overspeed setpoint was not changed.
21	Section 2.13.1 Risk Evaluation of Extended Power Uprate Page 285	Condensate flow recorder re-scaled	(delete)	There is no condensate flow recorder.
22	Pages 9, 256, 258, 279, 281, 283, 310, 342, 345	(see markup pages)	(as marked-up)	Typographical and editorial errors.

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Docket No. 50-271

Attachment 2

Vermont Yankee Nuclear Power Station

Extended Power Uprate

Clarifications to NRC's Final Safety Evaluation

Marked-up Pages

Total number of pages in Attachment 2 (excluding this cover sheet) is 29.

Grid Stability

The licensee's grid stability study identified several changes required for the grid to accept the uprated power. The modifications made were as follows:

- Increased the million volt-ampere (MVA) rating on the VYNPS - Northfield 345 kV line from 896 MVA to a minimum rating of 1075 MVA.
- Increased the MVA rating on the Ascutney-Coolidge 115 kV line from 205 MVA to 240 MVA.
- Added 60 MVA of shunt capacitors at the VYNPS 115 kV bus.
- Added a second primary protection scheme on the VYNPS north bus.
- Added a second primary protection scheme on the VYNPS main ~~generator~~. TRANSFORMER
- Replaced the VYNPS 381 breaker to provide independent pole tripping.
- Added out-of-step protection for the VYNPS generator.

Main Turbine - High Pressure Flow Path

The modifications associated with the main turbine high pressure flow path include replacement of the rotor and diaphragms; new control cams, camshafts, and hydraulics; new control valve settings, and turbine control and setpoint changes.

Instrumentation and Control Changes

The changes in various plant parameters at EPU conditions (e.g., steam flows) will require various instrumentation and control setpoint and calibration changes including the following:

- Electronic pressure regulator (mechanical hydraulic pressure control system for the turbine generator) setpoint change;
- Main steam line high flow setpoint change;
- Neutron monitoring setpoint changes (average power range monitor flow-biased scram and rod block monitors);
- Rod worth minimizer setpoint; and
- Turbine first stage pressure setpoint.

The NRC staff's evaluation of the licensee's plant modifications, within the scope of the areas of review, is provided in Section 2.0 of this SE.

EPU Review: EPU Review Standard RS-001, SE Section 2.3.5, "Station Blackout," requires that the NRC staff reach a conclusion that the licensee has adequately evaluated the effects of the proposed EPU on SBO and demonstrate that the plant will continue to meet the requirements of 10 CFR 50.63 following implementation of the proposed EPU. In order for the staff to reach this conclusion, Entergy needs to demonstrate that VYNPS meets the requirements in 10 CFR 50.63. The resolution of this issue is discussed in SE Section 2.3.5.

Appendix R Timeline for RCIC Initiation

Finding: The team identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," because from June 2001 to September 2004, the licensee did not adequately coordinate between the operations department and the engineering organization regarding procedure revisions that increased the length of time required to place the RCIC system in service from the alternate shutdown panels.

EPU Review: EPU Review Standard RS-001, SE Section 2.11, "Human Performance," requires the staff to conclude that the licensee has appropriately accounted for the effects of the proposed EPU on the available time for operator actions. The engineering inspection team found that the timeline for operator actions to place RCIC in service during an Appendix R scenario had been impacted due to procedure changes and that the licensee had not incorporated these changes into the VYNPS Safe Shutdown Capability Analysis (SCCA). However, the team found that at the current power level, during an Appendix R scenario, the operators have sufficient time to place RCIC in service from the alternate shutdown panels prior to reactor water level reaching the top of active fuel. At the proposed EPU power level, the team concluded that the margin was reduced such that the ability to place RCIC in service from the alternate shutdown panels prior to reactor water level reaching the top of active fuel was questionable. The resolution of this issue is discussed in SE Section 2.11.

SSCA

Periodic Testing of Motor-Operated Valves (MOVs)

Finding: The team identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," because the licensee had conducted MOV diagnostic tests using procedures that did not include acceptance limits, which were correlated to and based on applicable (stem thrust and torque) design documents. Additionally, MOV diagnostic testing had been conducted solely from the motor control centers using test instrumentation that had not been validated to ensure its adequacy.

EPU Review: EPU Review Standard RS-001, SE Section 2.2.4, "Safety-Related Valves and Pumps," requires that the NRC staff reach a conclusion that the licensee has adequately evaluated the effects of the proposed EPU on its MOV programs related to Generic Letters (GLs) 89-10, 96-05, and 95-07, and the lessons-learned from those programs for other safety-related power-operated valves. The engineering inspection team found that the licensee did not manage NRC commitments and conditions documented in the SE for the GL 96-05 MOV

generator reactive capability curves at EPU conditions maintain the generator stator core and field winding within their design limits, (i.e., no modification to the stator core and field winding is required for EPU). The generator hydrogen cooling system pressure is unchanged at EPU. However, the hydrogen cooling system heat exchangers have been replaced by heat exchangers of higher capacity due to increased heat removal requirements at EPU conditions. Additionally, the bushing current transformers (CTs) have been replaced for the EPU. ~~No generator protective relay changes are necessary, however, some protective relay setpoints will be modified for the rewound generator rating.~~

The NRC staff reviewed the licensee's submittals and concluded that the main generator would be acceptable for EPU based on the modifications described above.

2.3.2.3 Main Transformer

The main transformer is rated at 675 MVA. The main power transformer has recently been replaced and was sized to support the EPU. The associated switchyard components (rated for maximum transformer output) are adequate for transformer output. The loading on the main transformer is 650 MVA (main generator output of 684 MVA minus the 34 MVA house load fed through the unit auxiliary transformers), which is below the main transformer rating of 675 MVA.

The NRC staff reviewed the licensee's submittal and concluded that the main power transformer and the associated switchyard components are adequate for the uprated generator output and, therefore, operating the main transformer at the uprated power condition is acceptable.

2.3.2.4 Isophase Buses

The isophase bus duct connects the main generator to the primary windings of the main transformer and the unit auxiliary transformer and is rated at 17.9 kilo-amps (KA). The rating at the EPU conditions will be 19 KA. The NRC staff questioned the licensee regarding how the capacity of the isophase bus duct would be increased for the EPU. By letter dated January 31, 2004 (Reference 6), the licensee responded that the isophase bus duct is being upgraded from a rating of 17.9 KA to a rating of 19 KA by replacement of the bus duct cooler and by internal modifications to the bus duct cooling air distribution system.

The NRC staff reviewed the licensee's submittals and concluded that the operation of the isophase bus duct would be acceptable for the EPU after upgrading it from a rating 17.9 KA to a rating of 19 KA.

2.3.2.5 Unit Auxiliary Transformer (UAT)

The UAT is rated at 39.2 MVA. The EPU output is 34.4 MVA based on the worst-case loading.

The NRC staff reviewed the licensee's submittals and concluded that the increase in house loads resulting from the EPU is below the maximum UAT design rating and, therefore, operating the UAT at the uprated power condition is acceptable.

17.6

26.0

2.3.2.6 Startup Transformers

The two startup transformers are each rated at 28 MVA. Under EPU conditions, the loading on one transformer is ~~17.9~~ MVA and the loading on the other is ~~24.8~~ MVA.

The NRC staff reviewed the licensee's submittals and concluded that the startup transformers are not impacted by the EPU, and, therefore, operating the startup transformers at the uprated power condition is acceptable.

2.3.2.7 Station Loads

The licensee reviewed the station loads under normal, transient, and emergency operating scenarios for EPU conditions. In all cases, loads were computed based on equipment nameplate ratings or brake horsepower and were found acceptable for the EPU conditions. However, the licensee's application did not provide an evaluation for the operation of condensate and reactor feedwater pump motors at higher summer temperatures at the EPU conditions. In response to an NRC staff RAI, the licensee provided a detailed analyses for the condensate and reactor feedwater pump motors by letter dated May 19, 2004 (Reference 8). The two Westinghouse condensate pump motors and one GE condensate pump motor are adequate for operation at EPU conditions. Both the Westinghouse and the GE analyses bound the predicted pump flow run out. The Westinghouse reactor feedwater pump motors are adequate for operation at the EPU conditions. The feedwater pump motors remain adequate for all pump operating conditions including flow run-out.

The NRC staff reviewed the licensee's submittals and concluded that the station loads are not impacted by the EPU, and, therefore, operating the VYNPS with station loads at the uprated power condition is acceptable.

<u>Conclusion</u>

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the offsite power system and concludes that the offsite power system will continue to meet the requirements of draft GDC-39 following implementation of the proposed EPU. Results of these evaluations show that following implementation of the proposed modifications to the main generator, isophase bus duct and an addition of a 60 MVAR capacitor bank, the design will be acceptable for EPU conditions. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the offsite power system.

2.3.3 <u>AC Onsite Power System</u>

<u>Regulatory Evaluation</u>

The alternating current (ac) onsite power system includes those standby power sources, distribution systems, and auxiliary supporting systems provided to supply power to safety-related equipment. The NRC staff's review covered the descriptive information, analyses, and referenced documents for the ac onsite power system. The NRC's acceptance criteria for the ac onsite power system are based on draft GDC-24 and 39, insofar as they require the system to have the capacity and capability to perform its intended functions during

revised operating ranges of the affected process parameters at EPU conditions. Where necessary, the licensee revised the calibration and/or the setpoint and uncertainty calculations for the affected instruments. As discussed in Reference 6, there are no changes to instrument control philosophy as a result of EPU except for the new recirculation runback logic. That change is evaluated in SE Section 2.5.4.4. The proposed EPU does not change the safety functions or design basis of the VYNPS instrumentation with respect to separation, redundancy, or diversity.

The licensee's evaluation of the suitability of the existing instruments for EPU operation resulted in the following changes:

Parameter	Change
Main Steam Line (MSL) High Flow	Respan transmitters to cover new 140% steam flow value.
MSL High Flow	Replace 4 of the transmitters used to provide the 40% setpoint with more accurate transmitters. The setpoint remains at 40% of CLTP.
MSL High Flow	Setpoint changes for new setpoint for 140% isolation at new steam flows.
MSL High Flow	Install new indicators on the master trip units.
Neutron Monitoring	Average Power Range Monitor (APRM) flow-biased scram analytical limits (ALs) and rod block limits will be changed for the EPU.
Neutron Monitoring	APRM will be re-calibrated to reflect EPU operation.
Neutron Monitoring	Rod Block Monitors (RBM) will be re-calibrated to reflect EPU operation.
MSL Radiation Monitor	Normal setpoint changes based on new 100% MSL radiation level.
Feedwater Control (FWC) System, Feed Flow	Respan transmitters for EPU flows.
FWC System, Feed Flow	New indicator/recorder ranges for EPU flows.
FWC System, Steam Flow	Respan transmitters for EPU flows.
FWC System, Steam Flow	New indicator/recorder ranges for EPU flows.
Rod Worth Minimizer	Change the setpoint to maintain the setpoint at the same absolute value of steam flow because of the range changes of the associated instruments.
~~Recirculation Pump Net Positive Suction Head (NPSH) trip~~	~~Change setpoint to maintain the setpoint at the same absolute value of steam flow because of the range changes of the associated instruments.~~

<DELETE>

Header

Turbine First Stage Pressure	Setpoint change for the scram bypass.
Turbine Control System	Operating setpoint change to address increased steam line differential pressure.
Condensate Flow	Respan transmitters for EPU flows.
Condensate Flow	Computer point respan.
Condensate Heater Pressure Low	Setpoint change.
Condensate Flow to Oxygen Injection System	Instrument recalibration.
Steam Line Leak Alarm Module	Recalibration of transmitter and alarm module.
Condensate Pump Discharge Pressure	Indicator rebanding for new normal pressure.
Feedwater Pump Suction Pressure	Instrument recalibration.
Feed Pump Low Suction Pressure trip	Setpoint change for low-pressure pump trip.
Feed Pump Low Suction Pressure	Add a second pressure switch to each pump to provide signal for recirculation runback on loss of condensate pump.
Recirculation Motor Generator Control	New runback to reduce reactor power on loss of feedwater or condensate pump.

Since the instrumentation and control functions related to the above changes will be confirmed during post-modification testing, power ascension testing, instrument calibration, and TS surveillance testing, as applicable, the NRC staff has reasonable assurance that the instrumentation and related systems will continue to perform their intended safety functions at EPU conditions.

reactor feedwater pump trip

Conclusion

The NRC staff has reviewed the licensee's application related to the effects of the proposed EPU on the functional design of the reactor trip system, ESFAS, safe shutdown system, and control systems. The NRC staff concludes that the licensee has adequately addressed the effects of the proposed EPU on these systems and that the changes that are necessary to achieve the proposed EPU are consistent with the plant's design basis. The NRC staff further concludes that the systems will continue to meet the requirements of 10 CFR 50.55a(a)(1), 10 CFR 50.55(a)(h), and draft GDC-1, 11, 12, 14, 15, 19, 20, 22, 23, 25, 26, 40, and 42. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to instrumentation and controls.

and stuck open relief valve

Technical Evaluation

The licensee has evaluated the impact of the proposed EPU on the capability of the SWS (including the RHRSWS) to perform its safety functions (Reference 11, Attachment 1, response to RAI SPLB-A-8). Except for the SBO ~~and~~ Appendix R events, the licensee's analyses for EPU operation use the same SWS flow rates that are credited for the current licensed power level and therefore, no system modifications are required. As discussed in Reference 26 and in SE Section 2.6.5, the licensee originally credited one RHRSW pump for the SBO and Appendix R analyses. For the EPU, the licensee has revised these analyses to credit two RHRSW pumps.

Essential components that are serviced by the SWS include the RHR heat exchangers, SFPCS heat exchangers, EDG coolers, ECCS room coolers, and the RHRSWS pump motor coolers. The licensee indicated that key heat exchanger performance parameters that were used in the EPU analyses are consistent with the GL 89-13 (heat exchanger performance testing) program results. Entergy found that the suppression pool temperature and containment pressure will be higher when in the most limiting RHR suppression pool cooling and containment spray cooling modes; additional time will be required to cool down the reactor when in the shutdown cooling mode due to the higher reactor decay heat; and the ECCS corner room temperatures could increase by several degrees following a LOCA. However, because these effects do not cause any design limits of SSCs to be exceeded and because licensing-basis considerations will continue to be satisfied, the licensee concluded that the current SWS performance capability and flow balance are sufficient for EPU conditions. Note that GL 96-06 considerations do not apply to the SWS and are therefore not discussed in this section (see SE Section 2.5.3.3).

Regarding the SW temperature, the NRC staff raised a concern in reference to UFSAR Section 10.6.5, where it describes a higher temperature limit of 88°F under certain conditions, when the maximum design basis limit is 85°F. In response to RAI SPLB-A-15 (Reference 29, Attachment 1), the licensee stated that the higher SW temperature (i.e., 88°F) discussed in UFSAR Section 10.6.5 addresses a unique summer operating condition during hybrid mode of circulating water system operation which will not be applicable for EPU conditions. The revised design-basis analyses for EPU conditions assume a SW temperature limit of 85°F. The licensee stated that the UFSAR will be updated in conjunction with issuance of the EPU license amendment in this regard.

Based on a review of the information that was submitted, the NRC staff is satisfied that the licensee has adequately evaluated and addressed the impact of the proposed EPU on the capability of the SWS (including the RHRSWS) to perform its safety functions. Because design limits of SSCs will not be exceeded and licensing-basis considerations will continue to be satisfied, the staff agrees that the capabilities of the SWS will not be impacted by the proposed EPU. Furthermore, existing GL 89-13 programmatic controls will continue to assure that heat exchanger performance is maintained consistent with licensing-basis considerations following implementation of the proposed EPU.

the peak bundle power may increase slightly for EPU.

fuel cycle. The following sections address the effect of the EPU on fuel design performance and thermal limits.

Fuel Design and Operation

The PUSAR states that a CPPU increases the average power density proportional to the power increase and has some effects on operating flexibility, reactivity characteristics and energy requirements. ~~The peak bundle power will increase from 7.02 MWt before the EPU to 7.37 MWt after the EPU.~~ The power distribution in the core is changed to achieve increased core power, while limiting the Minimum Critical Power Ratio (MCPR), Linear Heat Generation Rate (LHGR), and Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) in any individual fuel bundle to be within its operating limits as defined in the core operating limits report (COLR).

As discussed in the NRC's SE for the CLTR (Reference 52), licensees using GE fuel, up through GE-14 fuel, may reference the CLTR as the basis for their EPU. As of RFO 25 (fall 2005), the VYNPS core utilizes GE-14 fuel only (Reference 33). Section 2.1 of the PUSAR states that [[

]] The fuel design limits are established for all new fuel product line designs as a part of the fuel introduction and reload analyses. [[

]]

The PUSAR further states that the percent power level above which fuel thermal margin monitoring is required may change with a CPPU. The original plant operating licenses set this monitoring threshold at a typical value of 25% of Rated Thermal Power (RTP). [[

]]

For a CPPU, the fuel thermal margin monitoring threshold is scaled down, if necessary, to ensure that monitoring is initiated [[

]], then the existing power threshold value must be lowered by a factor of $1.2/P_{25}$. The licensee stated that for VYNPS, the CPPU fuel thermal monitoring threshold is established at 23% of CPPU RTP. A change in the fuel thermal monitoring threshold also requires a corresponding change to the TS reactor core safety limit for reduced pressure or low core flow.

Because the licensee will continue to use approved analytical methods, and will continue to ensure that the results of those analyses remain within currently acceptable limits, the NRC staff finds the proposed EPU acceptable with respect to fuel design and operation.

Thermal Limits Assessment

The NRC's acceptance criteria require that the reactor core and the associated control and instrumentation systems be designed with appropriate margin to ensure that fuel design limits are not exceeded during normal operation, including AOOs. Operating limits are established to assure that regulatory or safety limits are not exceeded for a range of postulated events (transients and accidents).

The safety limit minimum critical power ratio (SLMCPR) ensures that 99.9% of the fuel rods are protected from boiling transition during normal operation and AOOs. The operating limit minimum critical power ratio (OLMCPR) assures that the SLMCPR will not be exceeded as the result of an AOO. NRC staff experience with several power uprates has shown that the change in OLMCPR resulting from a constant-pressure EPU is small. The CLTR SE (Reference 52) stated that this [[ ]] When the core design is complete, the OLMCPR will be determined with the "real" core design parameters. Because the licensee will use approved methods to evaluate these parameters, this is acceptable to the staff. As required by the CLTR SE, the licensee will perform [[ ]] to demonstrate that the SLMCPR and OLMCPR are appropriate for establishing the CPPU thermal limits.

The maximum average planar linear heat generation rate (MAPLHGR) operating limit is based on the most limiting LOCA conditions, and ensures compliance with the ECCS acceptance criteria in 10 CFR 50.46. For every new fuel type, GE performs LOCA analyses to confirm compliance with the LOCA acceptance criteria, and for every reload licensees confirm that the MAPLHGR operating limit for each reload fuel bundle design remains applicable.

In general, the licensee must ensure that plant operation is in compliance with the cycle-specific thermal limits ~~(SLMCPR, OLMCPR, MAPLHGR, and maximum LHGR)~~ and specify the thermal limits in a cycle-specific COLR as required by VYNPS TSs. In addition, while EPU operation may result in an increase in fuel burnup, the licensee cannot exceed the NRC-approved burnup limits. In accordance with VYNPS TS Section 6.6.C, cycle-specific analyses are performed using NRC-reviewed and-approved methodologies. The NRC staff finds that the licensee has appropriately considered the potential effects of EPU operation on the fuel design limits, and the generic thermal limits assessment show that the VYNPS can operate within the fuel design limits during steady state operation, AOOs, and accident conditions.

The TS 1.1.A requirements for SLMCPR assure that the fuel system is not damaged as a result of normal operation and AOOs. Compliance with 10 CFR 50.46, as discussed in SE

MCPR, APLHR, and LHGR

Conclusion

The NRC staff has reviewed the licensee's [[   ]] and concludes that it is consistent with the staff's understanding described in the CLTR SE. In addition, the licensee will perform plant-specific reload analyses to confirm that fuel design limits and RCPB pressure limits will not be exceeded under EPU conditions, and that the core design is not susceptible to thermal-hydraulic instability. Based on this, the NRC staff concludes that the thermal and hydraulic design will continue to meet the requirements of draft GDC-6 and 7 following implementation of the proposed EPU, and is acceptable to the staff.

2.8.4 Emergency Systems

2.8.4.1 Functional Design of Control Rod Drive System

Regulatory Evaluation

The NRC staff's review covered the functional performance of the control rod drive (CRD) system to confirm that the system can effect a safe shutdown, respond within acceptable limits during AOOs, and prevent ~~or mitigate the consequences of postulated accidents~~. The NRC's acceptance criteria are based on (1) ~~draft GDC-40 and 42, insofar as they require that protection be provided for ESFs against the dynamic effects and missiles that might result from plant equipment failures, as well as the effects of a LOCA; (2) draft GDC-26, insofar as it~~ requires that the protection system be designed to fail into a safe state; (3) draft GDC-31, insofar as it requires that the reactivity control systems be capable of sustaining any single malfunction without causing a reactivity transient which could result in exceeding acceptable fuel damage limits; (4) draft GDC-27 and 28, insofar as they require that at least two independent reactivity control systems be provided, with both systems capable of making and holding the core subcritical from any hot standby or hot operating condition, sufficiently fast to prevent exceeding acceptable fuel damage limits; (5) draft GDC-29 , insofar as it requires that at least one of the reactivity control systems be capable of making the core subcritical under any condition sufficiently fast to prevent exceeding acceptable fuel damage limits; (6) draft GDC-32, insofar as it requires that limits, which include considerable margin, be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the reactor coolant pressure boundary or (b) disrupt the core, its support structures, or other vessel internals sufficiently to impair the effectiveness of emergency core cooling; and (7) 10 CFR 50.62(c)(3), insofar as it requires that all BWRs have an alternate rod injection (ARI) system diverse from the reactor trip system, and that the ARI system have redundant scram air header exhaust valves. Specific review criteria are contained in SRP Section 4.6 and other guidance provided in Matrix 8 of RS-001.

subcritical under any conditions is unaffected by the proposed EPU. Control rod worth limits which include considerable margin are unaffected.

VYNPS has installed an alternate rod injection (ARI) system which is diverse from the reactor trip system, and the VYNPS ARI system has redundant scram air header exhaust valves.

Conclusion

The NRC staff has reviewed the licensee's plant-specific evaluation related to the effects of the proposed EPU on the functional design of the CRDS. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on the system and demonstrated that the system's ability to perform a safe shutdown, respond within acceptable limits, and prevent or mitigate the consequences of postulated accidents will be maintained following the implementation of the proposed EPU. The NRC staff further concludes that the licensee has demonstrated that sufficient technical basis exists to ensure the system's design bases will continue to be followed upon implementation of the proposed EPU.

The present design satisfies the draft GDCs under which the plant was licensed. No system changes are required for the EPU, so the system design will continue to meet draft GDCs and current licensing bases in this technical area. Based on this, the NRC staff concludes that the CRD system and ~~associated~~ analyses will continue to meet the requirements of draft GDC-26, 27, 28, 29, 31, 32, ~~40, and~~ 42, and 10 CFR 50.62(c)(3) following implementation of the proposed EPU. ~~Therefore, the~~ NRC staff finds the proposed EPU acceptable with respect to the functional design of the CRDS.

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2.8.4.2 Overpressure Protection During Power Operation

Regulatory Evaluation

Overpressure protection for the RCPB during power operation is provided by relief and safety valves and the reactor protection system. The NRC staff's review covered relief and safety valves on the main steam lines and piping from these valves to the suppression pool. The NRC's acceptance criteria are based on (1) draft GDC-9, insofar as it requires that the RCPB be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime; and (2) draft GDC-33, 34, and 35, insofar as they require that the RCPB be designed to assure that it behaves in a non-brittle manner and that the probability of rapidly propagating type failures is minimized. Specific review criteria are contained in SRP Section 5.2.2 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

Nuclear system pressure relief system is discussed in Section 4.4 of UFSAR. The safety/relief valves provide over-pressure protection for the nuclear steam supply system (NSSS), preventing failure of the nuclear system pressure boundary and an uncontrolled release of

frequency, extent and magnitude depend upon plant-specific parameters, valve locations, the valve design and piping support arrangements. The FIV of the piping will be addressed by the licensee by vibration testing during initial plant operation at the higher steam flow rates (Reference 16). Attachment 1 of Reference 16 describes the FIV testing during power ascension, and the Attachment 2 provides a regulatory commitment to implement this testing.

For the VYNPS over-pressure analysis with equilibrium core, the maximum calculated pressure meets the ASME Code. In addition, the most limiting pressurization transient is analyzed for each EPU reload cycle. Therefore, the NRC staff finds that the licensee has demonstrated an acceptable analysis of the plant response to over-pressure conditions for the proposed EPU. This provides reasonable assurance that the probability of gross rupture of the RCPB or significant leakage throughout its design lifetime will continue to be exceedingly low. Since the operating ranges of RPV pressure and temperature at the proposed EPU conditions remain unchanged, the RCPB design requirement to behave in a non-brittle manner to minimize rapidly propagating failures is unaffected.

Conclusion

The NRC staff has reviewed the licensee's plant-specific analyses with equilibrium core related to the effects of the proposed EPU on the overpressure protection capability of the plant during power operation. In addition, the licensee will perform plant-specific reload analyses to confirm that fuel design limits and RCPB pressure limits will not be exceeded under EPU conditions. Based on this, the NRC staff concludes that the overpressure protection features will continue to meet draft GDC-9, 33, 34, and 35 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to overpressure protection during power operation.

2.8.4.3 Reactor Core Isolation Cooling System

Regulatory Evaluation

The reactor core isolation cooling (RCIC) system serves as a standby source of cooling water to provide a limited decay heat removal capability whenever the main feedwater system is isolated from the reactor vessel. In addition, the RCIC system may provide decay heat removal necessary for coping with a station blackout. The water supply for the RCIC system comes from the condensate storage tank, with a secondary supply from the suppression pool. The NRC staff's review covered ~~the effect of the proposed EPU on the functional capability of the~~ system. The NRC's acceptance criteria ~~are~~ based on ~~(1) draft GDC-40 and 42, insofar as they require that protection be provided for ESFs against the dynamic effects that might result from plant equipment failures, as well as the effects of a LOCA; (2) draft GDC-37, insofar as it~~ requires that ESFs be provided to back up the safety provided by the core design, the RCPB, and their protective systems; (3) draft GDC-51 and 57, insofar as they require that piping systems penetrating containment be designed with appropriate features as necessary to protect from an accidental rupture outside containment and the capability to periodically test the

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NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on these events and demonstrated that the RCIC system will continue to provide sufficient decay heat removal and makeup for these events following implementation of the proposed EPU. Based on this, the NRC staff concludes that the RCIC system will continue to meet the requirements of draft GDC-37, ~~40, 42,~~ 51, and 57, and 10 CFR 50.63 following implementation of the proposed EPU. There~~fore, the~~ NRC staff finds the proposed EPU acceptable with respect to the RCIC system.

2.8.4.4 Residual Heat Removal System

Regulatory Evaluation

The RHR system is used to cool down the RCS following shutdown. The RHR system is typically a low pressure system which takes over the shutdown cooling function when the RCS pressure and temperature are reduced. The NRC staff's review covered the effect of the proposed EPU on the functional capability of the RHR system to cool the RCS following shutdown and provide decay heat removal. The NRC's acceptance criteria are based on draft GDC-40 and 42, insofar as they require that ESFs be protected against dynamic effects. Specific review criteria are contained in SRP Section 5.4.7 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

The RHR system is described in Section 4.8 of the VYNPS UFSAR. The RHR system is designed to (1) restore and maintain the reactor coolant inventory and to remove sensible and decay heat from the primary system and containment following reactor shutdown for both normal shutdown and post accident conditions. The RHR system is designed to operate in the low-pressure coolant injection (LPCI) mode, shutdown cooling (SDC) mode, suppression pool cooling (SPC) mode, containment spray cooling (CSC) mode and fuel pool cooling (FPC) assist mode. The LPCI mode, as it relates to the LOCA response, is discussed in Section 2.8.5.6.2 of this SE. The effects of the EPU on the other modes are described below. The results of the following evaluations are consistent with the generic evaluation in Section 4.1 of ELTR2.

The operational objective of normal shutdown is to reduce the bulk reactor temperature after scram to 125°F within approximately 20 hours using [[ ]] Single loop operation of RHR SDC is assumed for decay heat removal as part of the VYNPS Appendix R analysis in order to achieve cold shutdown within the time required by Appendix R (i.e., 72 hours). An underlying assumption in the Appendix R analysis is that one loop of RHR is unavailable due to the postulated event. The licensee's analysis shows that the time required to achieve cold shutdown (i.e., 212°F) under the Appendix R scenario conditions is less than 24 hours, and therefore, cold shutdown is achieved well within the 72-hour requirement assuming the operation of a single loop of RHR SDC. Since the SDC evaluation at EPU conditions demonstrated that the plant can meet this cooldown time, the NRC staff finds it acceptable.

The SPC and CSC modes of the RHR system cool the suppression pool following a design-basis LOCA by pumping the suppression pool water through the RHR heat exchangers and returning the water to the suppression pool, or by diverting the suppression pool water to spray headers in the drywell and wetwell after it has passed through the RHR heat exchangers. The effect of the proposed EPU with respect to these two RHR operating modes is discussed in Section 2.6 of this SE.

The FPC assist mode uses the RHR heat removal capacity to provide supplemental fuel pool cooling in the event that the fuel pool heat load exceeds the heat removal capacity of the fuel pool cooling and cleanup system. This mode can be operated separately or along with the fuel pool cooling and cleanup system to maintain the fuel pool temperature within acceptable limits. The effect of the proposed EPU with respect to FPC is discussed in Section 2.5.3 of this SE.

The licensee's application stated that the higher suppression pool temperature (194.7°F) and containment pressure during a postulated LOCA do not affect hardware capabilities of the RHR equipment, except for the RHR pump seals. The peak suppression pool temperature during a limiting LOCA remains below the RHR pump seal design temperature of 210°F. However, this temperature exceeds the maximum operating temperature of 185°F analyzed for the pump seals. In Reference 6, the licensee confirmed that the seals have been re-qualified for the increased suppression pool temperature under accident conditions.

197°F

Conclusion

The NRC staff has reviewed the licensee's plant-specific evaluation related to the effects of the proposed EPU on the RHR system. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on the system and demonstrated that the RHR system will maintain its ability to cool the RCS following shutdown and provide decay heat removal. Based on this, the NRC staff concludes that the RHR system will continue to meet the requirements of draft GDC-40 and 42 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the RHR system.

2.8.4.5 Standby Liquid Control System

Regulatory Evaluation

The standby liquid control system (SLCS) provides backup capability for reactivity control independent of the control rod system. The SLCS functions by injecting a boron solution into the reactor to effect shutdown. The NRC staff's review covered the effect of the proposed EPU on the functional capability of the system to deliver the required amount of boron solution into the reactor. The NRC's acceptance criteria are based on (1) draft GDC-27 and 28, insofar as they require that at least two independent reactivity control systems, preferably of different design principles, be provided, with both systems capable of making and holding the core subcritical from any hot standby or hot operating condition, sufficiently fast to prevent exceeding acceptable fuel damage limits; (2) draft GDC-29, insofar as it requires that at least one of the

Considering that the additional margin was obtained by a more conservative statistical treatment than currently used in the NRC-approved process, a 0.02 margin is considered to be a good SLMCPR margin. The code-to-code trending does not indicate degraded performance of the corrected TGBLA at high void conditions. Therefore, the staff accepts the 0.02 SLMCPR increase as sufficient in providing adequate margin, until the neutronic method is confirmed against appropriate measurement data.

2.8.7.2.2.3.2. LHGR Limit

The linear heat generation rate (LHGR) is a thermal-mechanical limit that assures the integrity of the fuel cladding during steady-state and transient conditions. During heat-up, a limit is placed on the peak pin nodal power to assure that the diametric strain would not result in [[ ]] (e.g., due to differential pellet/cladding creep and swelling). During a transient, the fuel pellet experiences overpower, which could result in fuel centerline melt. Therefore, a limit is also placed on the peak pin nodal power to prevent fuel centerline melting during any transient event. The peak kW/ft limit is exposure dependent and the thermal and mechanical limit establishes the steady-state kW/ft value. The peak kW/ft limit is an indicator of the peaking in the core since it comprises the combination of radial, axial, and local (pin) power peaking.

Margins in the operating LHGR kW/ft are of interest because the accuracy of the local pin peaking and the bundle power are contributors to the nodal pin kW/ft value. The table below shows the power/exposure dependent LHGR limit for GE-14 uranium dioxide (UO2) and gadolinium (Gd) rods. [[

GE-14 LHGR limits

[incorrect value]

			]]

In general, core monitoring operating data indicate that plants can operate with the peak pin at the LHGR limit for some limited amount of time. Therefore, any underpredictions in the nodal peak pin power peaking, the nodal bundle power and its operating history would translate to errors in the calculations of the operating pellet kW/ft with depletion. Peak rods in a bundle could be operating at the LHGR limit because of high bundle and pin power peaking. The peak

2.8.7-12 through 2.8.7-16

void branches only at 40% void history) against void reactivity coefficients calculated at different void ~~histories with~~ exposure. A plot showing the potential errors is presented in Figures ~~2.8.7-7 through 2.8.7-11.~~ These figures show that the fitting process results in increasing ~~errors with exposure~~ and burnup (as previously discussed, based on the void reactivity coefficient plots). The errors for exposures of about 30 GWd/t and less are consistent with the [[   ]] uncertainty assumed in the ODYN analysis. However, for higher exposures at high void fractions the error increases to more than 30%.

The GE approach to cross-section parameterization and fitting (quadratic fitting with values at 0%, 40%, and 70% void fraction) combined with the assumption that the change in cross sections with instantaneous change in voids is not sensitive to the void history, results in a substantially larger error in the void reactivity coefficient than assumed in the ODYN uncertainty analysis, particularly at high fuel exposures. The NRC staff determined that the impact of these increased errors on the response of the core during a transient needs to be evaluated to ensure that there is no impact on the core response. GE's position is that the errors at the higher exposures are not significant because the power generation in those bundles would be low. However, criteria applied to the fuel are also exposure dependent (e.g., LHGR), and therefore it is important to provide a demonstration that the fuel integrity is not compromised in the event of a transient event.

GE's Evaluation of Increased Void Coefficient Uncertainty

In response to NRC staff RAI-SRXB-A-68 (Reference 36), GE performed an evaluation of the errors in the void coefficient resulting from the cross section model as described above. The model assumptions that have a significant impact on the void coefficient are:

- The assumption that the cross sections can be parameterized with respect to void history using a quadratic fit to the 0%, 40%, and 70% instantaneous void fraction values with extrapolation to higher instantaneous void fractions. This results in a linear variation in coolant void reactivity with respect to void fraction, whereas the results show a significant deviation from the linear at high void fractions.

- The assumption that the void reactivity coefficient determined at a 40% void history condition applies to all other void histories. At high exposures the difference in isotopic compositions resulting from differing void histories results in significantly different void reactivity coefficients.

In the RAI response, GE considered the cross section model impacts separately for exposures less than 25 GWd/ST and greater than 25 GWd/ST, up to 65 GWd/ST. The calculation and comparison of the void coefficients at exposures of less than 25 GWd/ST indicated that the void reactivity coefficient errors were within those assumed in the ODYN ΔCPR/initial critical power ratio (ICPR) uncertainty analysis (see Figure 2.8.7-17). The results for exposures greater than 25 GWd/ST are shown in Figure 2.8.7-18 as a ratio of the MCNP to TGBLA06 void coefficients, and are quite similar to the confirmatory results discussed above. These relatively large

Kinf Comparisons

Regarding Kinf, the results agree very well at high exposures. The differences are larger in the 0 – 20 GWd/ST burnup range, which is when the gadolinium is burning out. The burnout of Gd, particularly at high void fractions, results in differences between TGBLA-6 and CASMO-4. The differences are smaller for void fractions of 40% and 70%, which represent the average void fraction value for a core. The differences for the results at high void fractions will not have a significant impact on the overall core results because the contribution of power in these regions will be relatively small.

Figures 2.8.7-19 and 2.8.7-20 provides two Kinf vs. exposure curves for historical void fractions of 40% and 70%. The calculations were performed with an instantaneous void fraction of 0%. Therefore, the difference at high exposure is caused by differing isotopic compositions resulting from the depletion at different void histories. As these figures show, the impact of void history at high exposures is significant (greater than 5% delta k). Confirmatory analyses performed with HELIOS, presented in Figure 2.8.7-21, show a similar trend and include data for a historical void fraction of 90%.

Pin Power Distribution Comparisons

The pin peaking factor (Figure 2.8.7-22) shows good agreement at zero exposure, when the power peaking is the most significant. Differences increase with exposure resulting in several percent at burnups of 60 GWd/ST. Generally, the TGBLA-6 local peaking factors are larger than CASMO-4 and therefore will be more conservative in regards to pin exposure and LHGR. Note that at high exposures (greater than 30 GWd/ST), the power peaking is the largest for bundles with the highest void history. However, given that the power generation in this highly burned fuel is low, the net result is that the overall LHGR (a product of the region average power and the local power peaking) will be lower than that earlier in the exposure history, where the differences are smaller. However, the LHGR limit is lower at higher fuel exposures. Therefore, if spectral shift strategy is employed, as proposed in the expanded operating domains for EPU plants, with the upper nodes depleting at high void conditions and a top-peaked power distribution, both the bundle power and the pin power peaking would be high, at the most limiting kW/ft exposures.

Figure 2.8.7-23 shows the root-mean-square (RMS) of the differences between TGBLA-6 and CASMO-4. The RMS ~~error is used in development of the SLMCPR.~~ Previous analysis comparing TGBLA-6 and MCNP (Reference 69, Enclosure 3) had shown that for a variety of lattices and exposures, that the RMS difference is [[ ]] However, in these comparisons the isotopic concentrations were taken from TGBLA and used in MCNP and, therefore, errors in depletion were not included. The results in this figure show that on a code-to-code basis that, for the lower void fractions, the RMS difference at low exposures and at high void fraction exceeds the [[ ]] RMS value, with maximum differences of about 2.5%. For the burnups above the gadolinium burnout and for lower void fractions, the differences are consistent with the [[ ]] value. Note that the results for the 90% void fraction presented in Figure 2.8.7-23 does not include potential errors caused by the quadratic fit and extrapolation used in GE's

vital areas access, at the proposed EPU power level of 1912 MWt. The NRC staff reviewed this design basis change and concludes that licensee continues to meet the applicable requirements.

Therefore, following implementation of the EPU, VYNPS will continue to meet its design basis in terms of radiation shielding, in accordance with the criteria in SRP Section 12.4, draft GDC 11, and NUREG-0737, item II.B.2.

2. Public and offsite radiation exposures.

There are two factors associated with the EPU that may impact public and offsite radiation exposures during plant operations. These are the possible increases in gaseous and liquid effluents released from the site, and the increase in direct radiation exposure from radioactive plant components and solid wastes stored onsite. As described above, the proposed EPU will result in a 20% increase in gaseous effluents released from the plant during operations. This increase is a minor contribution to the radiation exposure of the public. The nominal annual public dose from plant gaseous effluents for the VYNPS station is about 1 mrem. A 20% increase in this nominal dose is still well within the design criteria of 10 CFR Part 50, Appendix I.

The proposed EPU will also result in increased generation of liquid and solid radioactive waste. The increased condensate feed flow associated with the EPU results in faster loading of the condensate demineralizers. Similarly, the higher feed flow introduces more impurities into the reactor resulting in faster loading of the reactor water cleanup (RWCU) system demineralizers. Therefore, the demineralizers in both of these systems will require more frequent backwashing to maintain them. The licensee has estimated that these more frequent backwashes will increase the volume of liquid waste that will need processing by 1.2%, and an increase in processed solid radioactive waste by 17.8%. These increases are well within the processing capacity of the VYNPS radwaste system and are not expected to noticeably increase the liquid effluents or solid radioactive waste released from the plant. Therefore, these increases will have a negligible impact on occupational or public radiation exposure.

The most significant increase in offsite doses, from the proposed EPU will be due to increased N-16 skyshine and the direct exposure to radiation from miscellaneous radioactive waste stored on site. Based on measurements, the licensee has determined that the west boundary of the facility has the highest direct offsite radiation dose, nominally 15 mrem per year. The licensee has estimated that almost 90% of this dose, 13.4 mrem per year, is due to N-16 skyshine from the turbine building components. Skyshine is a physical phenomenon where gamma radiation that is released skyward during radioactive decay interacts with air molecules and, in this case, is scattered back down to the ground where it can expose members of the public. Since there is significantly less radiation shielding above the steam components in the turbine building, than there is to the sides of these components, skyshine from N-16 gamma radiation is a significant contributor to offsite dose rates. As discussed above, the licensee has estimated that plant operations at the EPU power level will increase the N-16 activity in the turbine building by 26%. Therefore, the gamma dose rate from N-16 skyshine at the west site boundary will likely

80%

2.11 Human Performance

2.11.1 Human Factors

Regulatory Evaluation

The area of human factors deals with programs, procedures, training, and plant design features related to operator performance during normal and accident conditions. The NRC staff's human factors evaluation was conducted to ensure that operator performance is not adversely affected as a result of system changes made to implement the proposed EPU. The NRC staff's review covered changes to operator actions, human-system interfaces, and procedures and training needed for the proposed EPU. The NRC's acceptance criteria for human factors are based on draft GDC-11, 10 CFR 50.120, 10 CFR Part 55, and the guidance in GL 82-33. Specific review criteria are contained in SRP Sections 13.2.1, 13.2.2, 13.5.2.1, and 18.0.

Technical Evaluation

Changes in Emergency and Abnormal Operating Procedures

The licensee indicated that the Emergency Operating Procedures (EOPs)/Severe Accident Management Guidelines (SAMGs) should remain unchanged in most aspects, with slight modifications required for some parameter thresholds and graphs which depend on the power and decay heat levels. These modifications would require changes in some values in the EOPs and the supporting documentation, but the adjustments would not affect the accident mitigation philosophy. Additionally, any change in scenario timings would be minor and would not significantly change the Human Error Probabilities (HEPs) in the risk assessments. The licensee will review the EOPs for any required changes, implementing those changes, and providing training to operators on the procedures.

For the Abnormal Operating Procedures (AOPs), the licensee indicated that some operator actions may be influenced by plant modifications required for supporting the increase in rated thermal power. The increased power level may require modifications to the AOPs and the supporting documentation. The licensee will review the AOPs to identify any effects of the EPU, including modifications to equipment and changes in setpoints to implement any changes to the AOPs, equipment, and setpoints necessary as a result of those effects, and to provide training to operators on the AOPs, equipment modifications, and setpoint changes.

Because no new procedures would be required, necessary changes to EOPs/SAMGs/AOPs, equipment and setpoints will be implemented, and training to address these changes will be provided, the NRC staff finds the licensee's proposed actions in this area to be acceptable.

Changes to Operator Actions Sensitive to EPU

The licensee stated that operator responses to transients, accidents, and special events would be minimally affected by EPU conditions. Operator actions for plant safety, after applicable

(or condensate pump)

automatic responses initiate, would not change as result of the EPU. The licensee's submittal described an operational enhancement that would provide for automatic recirculation runback given a single reactor feedwater pump (RFP) trip under EPU conditions, and this enhancement can be regarded essentially as the automation of an operator action.

The licensee also explained that there would be small reductions in time available for some operator actions due to the increase in decay heat for the EPU. Based on a screening process using risk assessment, the licensee identified a list of actions for explicit consideration. These actions were listed with current and EPU available times for completion. The licensee calculated the HEP for each action using industry standard techniques that included estimation of cognitive and manipulation times. The licensee stated that the shorter time limits would still be sufficient for operators to complete the tasks. To support the time estimates, the licensee performed interviews with multiple cognizant individuals, including a Senior Reactor Operator (SRO), a trainer, and an EOP developer. The time estimates were then entered into a HEP equation with standard deviation values based on an industry study of operator response times to over 100 different human actions. The licensee stated that differences in abilities of crews were taken into consideration in estimating the completion times and by using the HEP equation mentioned. Therefore, the licensee considered performance of all VYNPS operating crews to be bounded by the HEP calculations. Additionally, the licensee provided specific details on four of the most time-limited actions, which were all related to ATWS scenarios. For these actions, the licensee provided the current as well as EPU times available, along with estimated cognitive and manipulation times. For all four, the combined cognitive and manipulation estimated times were below the time limits under EPU.

As discussed in SE Section 1.6, an engineering inspection was performed at VYNPS, from August 9 through September 3, 2004, as documented in an inspection report dated December 2, 2004 (Reference 55). One of the inspection findings related to whether the licensee had appropriately accounted for the effects of the proposed EPU on the available time for operator actions related to a 10 CFR Part 50, Appendix R fire event. Specifically, the engineering inspection team found that the timeline for operator actions to place RCIC in service during an Appendix R scenario had been impacted due to procedure changes and that the licensee had not incorporated these changes into the VYNPS Safe Shutdown Capability Analysis (SCCA). However, the team found that at the current power level, during an Appendix R scenario, the operators have sufficient time to place RCIC in service from the alternate shutdown panels prior to reactor water level reaching the top of active fuel. At the proposed EPU power level, the team concluded that the margin was reduced such that the ability to place RCIC in service from the alternate shutdown panels prior to reactor water level reaching the top of active fuel was questionable. The specific details from the inspection report are as follows:

SSCA

> The Vermont Yankee SSCA relies on the reactor core isolation cooling (RCIC) system to be placed in service from the alternate shutdown panels prior to reactor water level reaching the top of active fuel following a loss of feedwater flow. In December 1999, the Vermont Yankee SSCA documented that, for the present day 100 percent power level, it would take 25.3 minutes for reactor water level to reach the top of active fuel following a

Operating Crew	Time to RCIC Initiation (minutes:seconds)
A	14:38
B	13:26
C	12:26
D	15:09
E	13:18
F	12:17
Average	13:32

SSCA

The licensee concluded that, based on the results of this demonstration, the assumption in the SGCA that the RCIC system can be made operable in approximately 15 minutes is confirmed. As discussed above, and as shown in PUSAR Table 6-5, the time to core uncovery calculated for EPU conditions is 21.3 minutes. Therefore, the NRC staff concludes that sufficient margin exists to allow operator action to manually start the RCIC system during an Appendix R event.

Based on the above discussion (i.e., automatic recirculation pump runback, ATWS scenarios, Appendix R event), the NRC staff concludes that there is reasonable assurance that the licensee has appropriately accounted for the effects of the proposed EPU on the available time for operator actions.

Changes to Control Room Controls, Displays and Alarms

The licensee analyzed potential system changes as result of the EPU and indicated that the following control room instrumentation would require modification: main steam line flow indicators would be replaced with digital units, feedwater (FW) flow indicators would be replaced with digital units, the main steam (MS) flow/FW flow recorder would be rescaled and use new chart paper~~, and the condensate flow recorder would be re-scaled~~. The licensee stated that none of these control room display changes would affect the Human Reliability Analysis results. Additionally, all modifications would be implemented in accordance with the VYNPS design modification process, which requires both human factors review and impact review by operations and training personnel. Training would also be provided on these changes.

The NRC staff finds that the licensee has adequately considered the equipment changes resulting from the EPU that affect operator ability to perform required functions.

- Adjustments to the VYNPS emergency operating procedures (EOPs) and severe accident management guidelines (SAMGs), to be consistent with CPPU operating conditions. The licensee stated that in almost all respects, the EOPs/SAMGs are expected to remain unchanged because they are symptom-based; however, certain parameter thresholds and graphs are dependent upon power and decay heat levels and will require slight modifications. EOP variables that play a role in the PSA and which may require adjustment for the EPU include:
  - Boron Injection Initiation Temperature
  - Heat Capacity Temperature Limit
  - Pressure Suppression Pressure Limit

- ~~Setpoint Changes~~
  - ~~Turbine overspeed~~
  - Turbine first-stage pressure steam scram bypass
  - Main steam line (MSL) high flow isolation

Initiating Event Frequencies

The VYNPS PSA addresses transients, loss of offsite power (LOOP), LOCAs, support system failures, internal floods, and external events.

The licensee stated that the proposed EPU is only expected to affect the frequency of the turbine trip (TT) initiating event. The frequency of this initiating event is affected because all three of the RFPs are required for power operation for the post-EPU condition. The licensee estimated that the TT frequency would increase by about 4% as a result of the proposed EPU. The NRC staff finds this change reasonable and concurs with the licensee's approach for adjusting the TT initiating event frequency.

The licensee stated that the frequency of total loss-of-feedwater (LOFW) is not expected to change as a result of the proposed EPU because failure of the RFPs is a negligible contributor to the overall frequency of this initiating event (total LOFW is dominated by other issues such as feedwater regulating valve closure). As part of the proposed EPU plant modifications, a reactor recirculation system runback modification will be installed to avoid a plant trip on loss of a condensate pump or RFP. The licensee stated that malfunction of the reactor recirculation system runback circuitry cannot cause a total loss-of-feedwater.

As a result of the proposed EPU, the plant's turbine bypass capacity will be reduced from 105% to 85% of rated steam flow. The licensee stated that the reduced capacity has no impact on the frequencies of transient initiating events because VYNPS does not use the large turbine bypass capacity to prevent a reactor trip given a load rejection event when reactor power is above approximately 30% of CLTP.

The licensee evaluated the impact of the proposed EPU on the LOOP frequency, and determined that there would be no impact. To confirm this conclusion, the NRC staff compared

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licensee considered three approaches to reflect the impact of the proposed EPU on the SORV probability:

- The upper bound approach would be to increase the SORV probability by a factor equal to the increase in reactor power (i.e., by a factor of 1.2 since the proposed EPU increases CLTP by 20%). This approach assumes that the SORV probability is linearly related to the number of SRV cycles, and that the number of cycles is linearly related to the reactor power increase.

- A less conservative approach is to assume that the SORV probability is linearly related to the number of SRV cycles. However, the number of cycles is not necessarily directly related to the reactor power increase. In this case, the postulated increase in SRV cycles due to the proposed EPU would be determined by thermal hydraulic calculations (e.g., ~~material access authorization program~~ (MAAP) runs).

- The lower bound approach would be to assume that the SORV probability is dominated by the initial cycle and that subsequent cycles have a much lower failure rate. In this approach, the pre-EPU SORV probability could be assumed to be insignificantly changed by a postulated increase in the number of SRV cycles.

The licensee applied the second approach to modify the SORV probability for the proposed EPU. The increase in the number of SRV cycles during accident response was estimated by comparing the results of MAAP runs for isolation transient scenarios performed in support of the post-EPU risk assessment. These analyses indicated that for the post-EPU plant, the number of SRV cycles in the first couple of hours of the accident progression increases by no more than 15%. Accordingly, the licensee increased the SORV probability by 15% in the post-EPU PSA model. The NRC staff agrees with the adjustment made by the licensee, and notes that the risk insights from the post-EPU PSA would not be expected to substantially change had the more conservative (first) approach been used.

The NRC staff finds that it is reasonable to conclude that equipment reliability will not change, as long as the operating ranges or limits of the equipment are not exceeded. For equipment that is operated within its operating ranges or limits, the staff notes that the licensee's component monitoring programs, as identified above, should detect significant degradation in performance and the staff expects these programs to maintain the current reliability of the equipment.

<u>Accident Sequence Delineation</u>

The success criteria for the VYNPS PSA are derived based on realistic evaluations of system capability over the 24-hour mission time of the PSA analysis. The licensee stated that approximately 60 Level I MAAP runs and 6 Level 2 MAAP runs were performed in support of the VYNPS post-EPU internal events Level 1 PSA.

amendment requests, the staff asked the licensee to provide a risk evaluation of the proposed containment accident pressure credit that addressed the five key principles of risk-informed decisionmaking contained in RG 1.174. The NRC staff also conducted a scoping risk evaluation to help confirm the licensee's risk evaluation.

The proposed containment accident pressure credit introduces uncertainty about the success criteria used to construct the PSA model. As discussed in SE Section 2.6.5, the available evidence strongly suggests that no containment accident pressure credit is required when realistic initial conditions and parameters are used to determine the available NPSH to the ECCS pumps. However, these initial conditions and parameters contain both aleatory uncertainties (e.g., changes in service water temperature caused by seasonal variations) and epistemic uncertainties (e.g., uncertainty in determining frictional head losses for a given piping configuration). In order to assess the impact of these uncertainties on risk, the licensee performed a sensitivity analysis (Reference 39, 40, and 44) of the proposed containment accident pressure credit. Specifically, the licensee modified its PSA model by assuming that the proposed containment accident pressure credit is needed and compared the results of this modified PSA model to the results of the post-EPU risk evaluation, which assumes that containment accident pressure credit is not required. The NRC staff used an identical approach in conducting its scoping risk evaluation. Section 2.2.5.5 of RG 1.174 states that sensitivity analyses may be used to assess the impact of modeling uncertainties.

The licensee's sensitivity analysis was based on the PSA model for internal events (including internal flooding). In the sensitivity analysis, it was assumed that a core-damage accident would occur if all of the following events occur:

- An accident occurs that discharges reactor coolant into the containment, adding heat to the suppression pool. The accident may discharge reactor coolant to the suppression pool either directly (e.g., a LOCA) or indirectly (e.g., a transient followed by a subsequent safety relief valve (SRV) opening). coolant LPCI
- The low-pressure safety injection (LPSI) or core spray (CS) pumps are needed to provide reactor inventory control or decay heat removal.
- Containment integrity is lost so that the containment accident pressure is not sufficient to provide adequate NPSH to the LPSI and CS pumps.
- The operator does not initiate alternative injection sources to provide core cooling.

The licensee's sensitivity analysis considered two specific containment failure modes: pre-existing, undetected leaks through the containment and failure of the primary containment isolation system. The probability of containment leaks was estimated using EPRI Report TR-1009325 (Reference 48). This information source, which used expert elicitation to develop containment leakage probabilities, is currently under NRC staff review as a technical basis for extending Type A integrated leak rate test (ILRT) intervals up to 15 years.

- An accident occurs that discharges reactor coolant into the containment, adding heat to the suppression pool. The accident may discharge reactor coolant to the suppression pool either directly (e.g., a LOCA) or indirectly (e.g., a transient followed by a subsequent SRV opening).

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- The LPSI or CS pumps are needed to provide reactor inventory control or decay heat removal.

- Containment integrity is lost so that the containment accident pressure is not sufficient to provide adequate NPSH to the LPSI and CS pumps.

- The operator does not initiate suppression pool cooling within 4 hours after the accident occurs.

Modifications to the interfacing system LOCA event trees were not needed because the containment is not pressurized following these types of events (leakage from the reactor coolant system is outside of the containment). Therefore, no heat is added to the suppression pool from interfacing system LOCAs and there is no need to ensure containment integrity to provide adequate NPSH to the ECCS pumps.

The NRC staff observes that a loss of containment integrity either prior to the accident (e.g., due to a pre-existing and undetected containment leak) or immediately after the accident (e.g., due to failure of the primary containment isolation system) will not cause an immediate failure of the LPSI and CS pumps because it takes time for the discharge of reactor coolant to sufficiently heat the suppression pool to the point where these pumps will cavitate. The licensee provided (Reference 24) a MAAP calculation that indicates the operator will have about 4 hours from the start of a large-LOCA to initiate suppression pool cooling and avoid pump cavitation.

The NRC staff's scoping risk evaluation considered three specific containment failure modes, whose probabilities were estimated as follows:

- Pre-existing, undetected leaks through the containment: the probability of this failure mode was obtained from the licensee's evaluation of the risk impacts of extending, on a one-time basis, the ILRT to 15 years (Reference 65). The NRC staff approved this ILRT extension on August 31, 2005 (Reference 66).

- Failure of the primary containment isolation system: the probability of this failure mode was obtained from the licensee's evaluation of the risk impacts of extending, on a one-time basis, the ILRT to 15 years as noted above.

- Failure of MSIVs to close on demand: the probability of this failure mode, including common-cause failures, was estimated using data obtained from RES.

The NRC staff's scoping risk evaluation indicated that the change in CDF associated with crediting containment accident pressure to provide adequate NPSH to the ECCS pumps was

With respect to accident sequence modeling (including event tree and systems analysis), the licensee stated that the proposed EPU does not change the plant configuration and operation in a manner such that new accident sequences or changes to existing accident scenario progressions result. The NRC staff observes that this conclusion is reasonable, given the changes to accident sequence success criteria identified by the licensee. The staff believes that the licensee could have utilized the PSA to assess the risk impacts of crediting containment accident pressure to provide NPSH to the ECCS pumps, and notes that doing so would have necessitated changes to the PSA logic model. Had this been a risk-informed application under RG 1.174, the staff would have pursued this matter further with the licensee in order to ensure that the post-EPU PSA was used to assess the risk impacts of the proposed containment accident pressure credit. However, as previously discussed, the staff has concluded that crediting containment accident pressure at VYNPS to provide NPSH to the ECCS pumps does not create "special circumstances" that rebut the presumption of adequate protection provided by the licensee meeting the currently specified regulatory requirements. Therefore, the NRC staff concludes that the accident sequence modeling used by the licensee in the post-EPU PSA is technically adequate to support the EPU application. The staff expects the licensee to make appropriate changes to the PSA model as required to address the risk impact of crediting containment accident pressure before submitting any future risk-informed license amendment requests under RG 1.174.

<u>Operator Actions and LOOP Recovery</u>

The VYNPS risk profile is dependent on the operating crew's actions for successful accident mitigation. The success of these actions is in turn dependent on a number of performance shaping factors. The performance shaping factor that is principally influenced by the proposed EPU is the time available to detect, diagnose, and perform required actions. The higher power level results in reduced times available for some actions. To quantify the potential effect of this performance-shaping factor, deterministic thermal-hydraulic calculations using the MAAP computer code were used. The licensee also examined the impact of the proposed EPU on the man-machine interface performance shaping factor. Changes to be made to the control room displays for the proposed EPU are:

- MSL flow indicators replaced with digital units
- Feedwater (FW) flow indicators replaced with digital units
- ~~Steam/FW flow recorder re-scaled~~
- Conden~~sate flow record~~er re-scaled

The licensee stated that none of these control room display changes affect the quantification of human error probabilities (HEPs) in the VYNPS PSA.

Not all operator actions in the VYNPS PSA have a significant effect on the results. The licensee performed a screening analysis to identify those operator actions that have a significant effect on the PSA results. The licensee's screening process was performed against the following criteria:

No.	COMMITMENT	REFERENCE (Entergy letter number and date)	TYPE		SCHEDULED COMPLETION DATE	NRC COMMENTS
			One-time	Continuing		
16	Implement flow induced vibration and steam dryer monitoring, including associated evaluation as necessary during EPU power ascension testing as described in Entergy letter BVY 04-100.	BVY 04-100 September 23, 2004 (Reference 16)	x		During EPU power ascension testing	
17	Discuss details of the acoustic analysis and steam dryer power ascension test acceptance criteria at a meeting with NRC staff.	BVY 04-100 September 23, 2004 (Reference 16)	x		September 30, 2004	Commitment was satisfied by meeting held on September 29, 2004.
18	Implement BWROG operational (moisture carryover) response guidance.	BVY 04-100 September 23, 2004 (Reference 16)	x		During EPU power ascension testing	
19	Revise the MOV Periodic Verification Program to include periodic at-the-valve testing and formalize the process for DC motor trending.	BVY 04-101 September 30, 2004 (Reference 17)	x		December 1, 2004	Attachment 3 to Entergy letter BVY 05-083, dated September 10, 2005 (Reference 33), states that this commitment is complete.
20	Verify the RCIC start time assumed in the SCCA and complete training of operations crews on the revised procedure.	BVY 04-107 September 30, 2004 (Reference 18)	x		December 1, 2004	Entergy letter BVY 04-131, dated December 8, 2004 (Reference 23), states that this commitment is complete.

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ACRONYM	DEFINITION
~~LPSI~~ LPCI	low pressure safety injection
LPZ	low population zone
LTR	licensing topical report
LWMS	liquid waste management system
MAAP	material ~~access authorization~~ program
MAPLHGR	maximum average planar linear heat generation rate
MBTU/hr	million British thermal units per hour
MCC	motor control center
MCES	main condenser evacuation system
MCS	main condenser system
MELLLA	maximum extended load line limit analysis
mg	milligram
Mlb/ft^2	million pounds per square foot
MOV	motor-operated valve
MS	main steam
MSIV	main steam isolation valve
MSL	main steam line
MSLB	main steam line break
MSSS	main steam supply system
MTU	metric ton uranium
MVA	megavolt amperes
MVAR	megavolt amperes reactive
MWe	megawatts electric
MWH	megawatt hours
MWt	megawatts thermal
n/cm^2	neutrons per centimeter squared

ACRONYM	DEFINITION
RTP	rated thermal power
RVFW	reactor vessel feedwater
RWCS	reactor water cleanup system
RWCU	reactor water cleanup
RWE	rod withdrawal error
RWM	rod worth minimizer
SAGs	severe accident guidelines
SAMG	severe accident management guidelines
SBO	station blackout
SCC	stress corrosion cracking
SSCA ~~SCCA~~	Safe Shutdown Capability Analysis
SDC	shutdown cooling
SDM	shutdown margin
SDMP	Steam Dryer Monitoring Plan
SDP	significance determination process
SE	Safety Evaluation
SFP	spent fuel pool
SFPAVS	spent fuel pool area ventilation system
SFPCCS	spent fuel pool cooling and cleanup system
SFPCS	standby fuel pool cooling system
SGTS	standby gas treatment system
SIL	Services Information Letter
SL	safety limit
SLC	standby liquid control
SLCS	standby liquid control system
SLMCPR	safety limit minimum critical power ratio