RAS 12256

DOCKETED
USNRC

2006 SEP 19 PM 3: 35

OFFICE OF THE SECRETARY
ADJUDICATIONS STAFF

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

U.S. NUCLEAR REGULATORY COMMISSION
In the Matter of Entergy Nuclear Vermont Yankee L.L.C.
Docket No. 50-271 Official Exhibit No. Entergy 6 (Staff 8)
OFFERED by: Applicant/Licensee Intervenor ________
NRC Staff Other ________
IDENTIFIED on 9/13/06 Witness/Panel Nichols/Casillas
Action Taken: ADMITTED REJECTED WITHDRAWN
Reporter/Clerk MAC

Template=SECY-028

SECY-02

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 ±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.