General Information or Other (PAR) Event # 41547

Rep Org:	GENERAL ELECTRIC NUCLEAR ENERGY	Notification Date / Time:	03/29/2005 19:17 (EST)
Supplier:	GENERAL ELECTRIC NUCLEAR ENERGY	Event Date / Time:	01/28/2005 (EST)
		Last Modification:	03/29/2005
Region:	1	Docket #:	
City:	WILMINGTON	Agreement State:	Yes
County:		License #:	
State:	NC		

NRC Notified by:	JASON S. POST	Notifications: JAMES TRAPP	R1
HQ Ops Officer:	STEVE SANDIN	MALCOLM WIDMANN	R2
Emergency Class:	NON EMERGENCY	KENNETH RIEMER	R3
10 CFR Section:		GREG PICK	R4
21.21	UNSPECIFIED PARAGRAPH	OMID TABATABAI	NRR

PART 21 REPORT INVOLVING POTENTIAL TO EXCEED LOW PRESSURE TECHNICAL SPECIFICATION SAFETY LIMIT

The following is a portion of a facsimile submitted by GE Energy-Nuclear:

"The defect is a calculation of an anticipated operational occurrence (AOO), which predicts that the Pressure Regulator Failure Maximum Demand (Open) (PRFO) transient will be terminated by a high water level trip as a result of level swell in the reactor. An improved (and approved) model predicts that MSIV closure will occur when steam line pressure reaches the low-pressure isolation setpoint (LPIS), rather than terminate due to a high water level trip. Depending upon the plant-specific response to a PRFO, including the value of the LPIS, reactor steam dome pressure could decrease to below 785 psig while thermal power exceeds 25% of rated, which would be a violation of SL 2.1.1.1. This constitutes a Defect as defined in 10 CFR 21.3, even though there is no safety hazard created. SL 2.1.1.1 was intended to protect fuel cladding integrity during startup conditions without the need to perform a Critical Power Ratio (CPR) calculation. The AOO of concern is a transient from normal operating conditions that causes CPR to increase, so the event produces additional margin to the Minimum Critical Power Ratio Safety Limit (SLMCPR) and does not threaten fuel cladding integrity. The LPIS should not be considered as a Limiting Safety System Setting (LSSS) for SL 2.1.1.1 since it does not provide a 'significant safety function' with regard to protecting fuel cladding integrity. This indicates that SL 2.1.1.1 is overly conservative because an event that causes CPR to increase and does not threaten fuel cladding integrity, may result in exceeding a reactor core SL."

Notified Plants:

AFFECTED: Nine Mile Point 2, Fermi 2, Pilgrim, Vermont Yankee, Limerick 1 & 2, Peach Bottom 2 & 3, Perry 1, and Hope Creek.

POTENTIALLY AFFECTED: Clinton, Oyster Creek, Brunswick 1 & 2, Nine Mile Point 1, FitzPatrick, Grand Gulf, River Bend, Dresden 2 & 3, LaSalle 1 & 2, Quad Cities 1 & 2, Cooper, Duane Arnold, Monticello, Hatch 1 & 2,

IE19

General Information or Other (PAR) Event # 41547
Browns Ferry 1, 2 & 3.
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GE Energy-Nuclear

General Electric Company
3901 Castle Hayne Rd., Wilmington, NC 28401

March 29, 2005
MFN 05-021

Document Control Desk
United States Nuclear Regulatory Commission
One White Flint North
11555 Rockville Pike
Rockville, Maryland 20852-2738

Subject: **10CFR21 Reportable Condition Notification: Potential to Exceed Low Pressure Technical Specification Safety Limit**

This letter provides information concerning a condition that GE has determined to be reportable under 10CFR21, even though it does not produce a substantial safety hazard. As a result of improvements in calculation methods, GE has identified an anticipated operational occurrence (AOO), the Pressure Regulator Failure Maximum Demand (Open) (PRFO) transient, that could result in a condition in which Safety Limit (SL) 2.1.1.1 may be exceeded. Depending upon the plant-specific response to a PRFO, reactor steam dome pressure could decrease to below 785 psig while thermal power exceeds 25% of rated, which would be a violation of SL 2.1.1.1. However, as provided in the technical specification bases, a Minimum Critical Power Ratio (MCPR) limit is established to ensure that reactor core safety limits is to protect fuel cladding integrity. A PRFO would result in an increase in the critical power ratio (CPR), thereby protecting fuel cladding integrity. This indicates that SL 2.1.1.1 is overly conservative as applied to a PRFO, because such an event would cause the CPR to increase and would not threaten fuel cladding integrity. This is explained in further detail herein.

Attachment 1 lists the affected and potentially affected plants as determined by GE records. Licensees may have additional evidence to support why a plant identified by GE as affected may be potentially affected, or a plant identified by GE as potentially affected may be unaffected. Attachment 2 provides information required for written notification to the NRC per §21.21(d).

Summary:

GE has continued to improve the methodology used for licensing basis transient analyses. The approved model has evolved from REDY, to ODYN, to TRACG. Reactor depressurization transients, such as Pressure Regulator Failure-Maximum Demand (Open) (PRFO), are non-limiting for fuel cladding integrity because Critical Power Ratio (CPR) increases during the event, and they are not typically included in the scope of reload evaluations. Recent investigations by GE have determined that even though

REDY, ODYN, and TRACG all show that the PRFO is a non-limiting transient since CPR increasing during the transient, the difference in reactor level swell predicted by REDY, vs. ODYN and TRACG, can impact the predicted plant response to the PRFO.

Technical Specification (TS) Safety Limits (SL) are specified to ensure that acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). Reactor Core SLs are set such that fuel cladding integrity is maintained and no significant fuel damage is calculated to occur if the SLs are not exceeded. The standard Improved Technical Specifications (ITS) specify SL 2.1.1.1 to require that thermal power shall be ≤ [25]% rated (the value is a plant-specific number), when reactor steam dome pressure is < 785 psig or core flow is < 10% of rated. Many plants have implemented the ITS or have TS that contain a similar SL. This SL was introduced to preclude the need for CPR calculations when reactor steam dome pressure is less than 785 psig. The power value in ITS SL 2.1.1.1 is selected to ensure that power remains well below the fuel assembly critical power for the conditions in which CPR calculations are not performed.

Previous evaluations by GE using the REDY model predicted that reactor water level would swell during a PRFO transient; the depressurization would be terminated by a high level turbine trip. However, level swell is difficult to predict and the level swell portion of transient models have larger uncertainties than other portions of the transient models. Recent evaluations by GE with the improved transient models have determined that the reactor level swell may not be sufficient to reach the high level trip, in which case the depressurization could be terminated by MSIV closure at the low-pressure isolation setpoint (LPIS). Depending upon the plant-specific response to a PRFO, including the value of the LPIS, reactor steam dome pressure could decrease to below 785 psig for a few seconds while thermal power exceeds 25% of rated, which would exceed the conditions in ITS SL 2.1.1.1. This indicates that ITS SL 2.1.1.1 is overly conservative, because an event that causes CPR to increase and does not threaten fuel cladding integrity can result in exceeding this reactor core SL.

Absent a plant-specific analysis that demonstrates otherwise, GE is identifying plants that have the potential for reactor dome pressure to drop below 785 psig for a PRFO as those with a LPIS-Analytical Limit (AL) of less than 785 psig. These plants are listed as Affected in Attachment 1. Sufficient evaluations to determine the plant specific conditions (for example, off-rated conditions) for which the PRFO will not cause reactor dome pressure to drop below 785 psig have not been completed and are judged to not be necessary since this is a non-limiting transient for challenges to fuel cladding integrity. Therefore, all other plants identified in Attachment 1 are considered to be potentially vulnerable to this condition.

The 10 CFR 21.3 definition of a *Defect* includes, "a condition or circumstance involving a basic component that could contribute to the exceeding of a safety limit, as defined in the technical specifications." Section 21.21(d) requires that the NRC be notified upon determination of the existence of a defect. Even though this condition does not threaten fuel cladding integrity or produce a significant safety hazard, it is defined as a *Defect* under 10 CFR 21 because the conditions in plant TS corresponding to ITS SL 2.1.1.1 may be exceeded, which requires NRC notification as a Reportable Condition.

10 CFR 50.36(c)(1) provides requirements for Limiting Safety System Settings (LSSS). Subparagraph (ii)(A) states, "*LSSS for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. Where a LSSS is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded.*" Plants may have identified the LPIS as a LSSS. However, in this instance, the LPIS does not have a "significant safety function" since its operation is not required to protect fuel cladding integrity, and so the LPIS should not be specified as a LSSS.

TS Bases for the main steam line (MSL) low pressure isolation state that MSIV closure ensures that the RPV temperature change limit of 100°F/hr is not reached, and that it supports actions to ensure SL 2.1.1.1 is not exceeded. The TS Bases states clearly that the LPIS-Allowable Value is based on preventing excessive RPV depressurization. While low-pressure MSIV closure supports actions to ensure SL 2.1.1.1 is not exceeded, this is not a "significant safety function" of the LPIS.

Background

Most operating BWRs have a TS SL that requires the core thermal power to be < [25]% of rated when reactor steam dome pressure is < 785 psig or core flow is < 10% of rated. This was intended to provide fuel cladding integrity protection during start-up conditions since the GEXL correlation (used by GE to perform CPR calculations) is not approved as a licensing model for pressures < 785 psig (~800 psia). GE has tested some fuel designs over an extended pressure range, in some cases as low as 600 psia. The critical power continues to increase monotonically as pressure decreases below the bottom of the approved range, and results in CPR increasing as pressure decreases. However, the GEXL correlation is not approved by the NRC for licensing calculations for pressures below 800 psia.

Licensees are required to demonstrate that no AOO will cause a SL to be exceeded, as documented in plant Safety Analysis Reports. AOOs that cause reactor depressurization are not routinely evaluated because thermal margins increase during these events, and therefore, they are non-limiting. The PRFO is the only AOO capable of depressurizing the reactor vessel enough to reach the MSIV low-pressure isolation setpoint (LPIS). PRFO evaluations with REDY showed that vessel level swell caused by the depressurization resulted in a high water level turbine trip, which resulted in the event being terminated before the pressure in the main steam line (MSL) reached the LPIS. Therefore, GE did not perform an evaluation of the PRFO or specify a LPIS-AL to demonstrate compliance with the conditions of ITS SL 2.1.1.1.

Evaluation

An evaluation of the PRFO with ODYN shows that the vessel level swell may not be sufficient to cause a high water level trip and the depressurization transient may not be terminated until the MSIV closes due to the low MSL pressure. The scenario for this event, is (1) a pressure regulator failure results in the maximum steam demand (specified by the value of the Maximum Combined Flow Limiter (MCFL), typically between 110 and 130%), (2) the turbine control valves (TCV) and the turbine bypass valves (TBV) open as required to meet the steam demand, (3) the steam flow increases and the pressure in the reactor steam dome and MSL decreases and (4) once the turbine inlet pressure (Pturb) reaches the LPIS, a MSIV trip occurs. The MSIV closure terminates the depressurization and scrams the reactor.

GE analysis assumes that the MSIV is tripped at the turbine inlet pressure corresponding to the LPIS-Analytical Limit (AL). The trip may occur significantly sooner since the Allowable Value and Nominal Trip setpoint implemented by the licensees may significantly exceed the LPIS-AL.

A typical PRFO response for a LPIS-AL of 720 psig is shown in Figure 1. Reactor dome pressure decreases until terminated by MSIV closure. This results in dome pressure (Pdome) dropping below the SL 2.1.1.1 value (785 psig = ~800 psia on Figure 1), while heat flux, which is indicative of thermal power, is still in excess of 25% of rated power. The CPR (not shown) continues to increase during the depressurization, so that the initial CPR is the limiting CPR condition during the entire transient. The MSIV closure signal is generated when the turbine inlet pressure (Pturb) reaches the LPIS-AL (shown as 735 psia in Figure 1). The conditions that exceed SL 2.1.1.1 exist for only a few seconds, and as stated previously, CPR increases during the event relative to the initial CPR value, so fuel cladding integrity is not threatened. Nonetheless, this is now a known AOO that could contribute to the exceeding of a safety limit, as defined in the technical specifications.

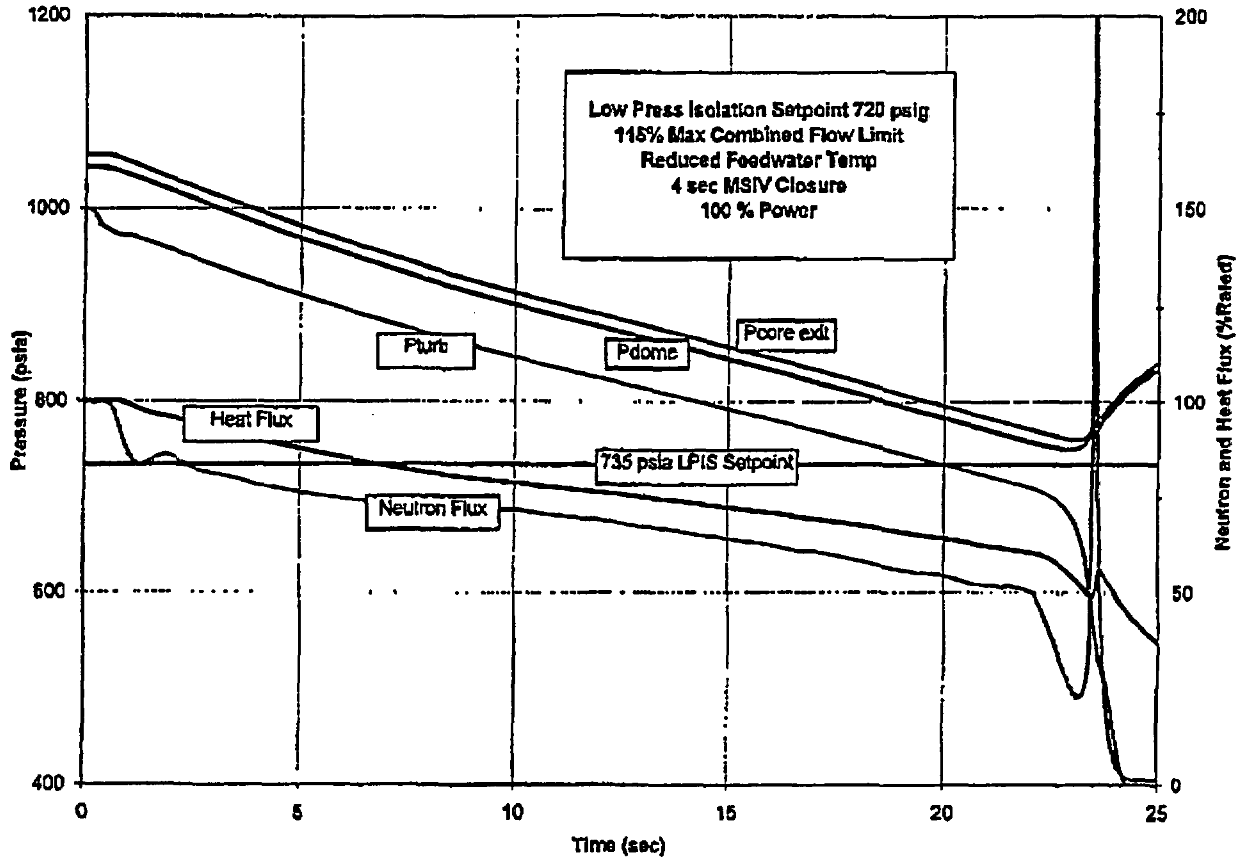

Figure 1. Typical PRFO Response

Important parameters which impact plant response to a PRFO event include (1) LPIS, (2) MCFL, (3) feedwater temperature, (4) steam line pressure drop, (5) MSIV closure time, (6) turbine inlet pressure sensor delay, and (7) initial core power. Table 1 lists values contained in GE records for the LPIS-AL (which is a turbine inlet pressure value) and MCFL. Actual plant-specific values may deviate from the values contained in GE records. The MCFL, feedwater temperature and initial core power impact the depressurization rate. The MCFL impacts depressurization rate because it determines the TCV and TBV opening. The reduced feedwater temperature and initial power less than rated are conditions that have less than rated steam flow. A PRFO from these conditions results in a larger steam flow mismatch than if initiated from rated conditions. This results because the TCV and TBV are assumed to open to the maximum demand regardless of initial conditions. The LPIS-AL determines the turbine inlet pressure at which the isolation is initiated. The steam line pressure drop maintains the reactor steam dome pressure above the turbine inlet pressure. The steam line pressure drop ranges from approximately 30 to 100 psi. The MSIV closure time and the turbine inlet pressure sensor delay determine how long the depressurization continues after the LPIS-AL has been reached.

The ranges (approximate) in these parameters is large; 720 to 850 psig for LPIS-AL, 108 to 130% for MCFL, 0 to 170°F for feedwater temperature reduction (most plants fall in the range of 70 to 105°F), 30 to 100 psi for steam line pressure drop, MSIV closure time is typically 3 sec (minimum) and 5 sec (maximum), and variation on the turbine inlet pressure sensor delay is unknown. The potential for operation at off-rated core powers must also be considered.

Several evaluations were performed within the range of parameters identified above, including off-rated initial operating conditions, to determine the possibility that the dome pressure would drop below the SL 2.1.1.1 value of 785 psig while thermal power exceeded [25]% of rated. From this limited set of evaluations, it was apparent the plants with LPIS-AL below 785 psig have the potential to experience a PRFO that results in the dome pressure dropping below 785 psig while thermal power exceeds [25]% of rated. Evaluations of the PRFO over a wide range of conditions led to the conclusion that all plants are considered potentially vulnerable to exceeding SL 2.1.1.1. While GE has not evaluated every possible plant combination to determine the full extent of which plants are/are not affected by this condition, GE believes that the evaluation is sufficient in this instance since the recommendation is that no compensatory actions are appropriate, and that SL 2.1.1.1 should be corrected so that it provides the intended fuel cladding integrity protection for startup conditions and does not provide unnecessary and overly conservative restrictions for events that do not threaten fuel cladding integrity.

Safety Basis

Technical Specification SL 2.1.1.1 requires that reactor power shall be ≤ [25]% of rated when reactor steam dome pressure is < 785 psig or core flow is < 10% of rated. SLs must be met for both normal operation and for anticipated operational occurrences (AOO). A Pressure Regulator Failure – Maximum Demand (Open) (PRFO) is defined as an AOO and an analysis using GE transient analysis methods and licensing basis assumptions has demonstrated that at least for some plants, this event can cause SL 2.1.1.1 to be exceeded, and is therefore, a reportable condition under 10 CFR 21. GE is not aware of any AOO that has caused SL 2.1.1.1 to be exceeded, and plant-specific MSIV low-pressure isolation setpoints that are higher than the LPIS-AL make it less likely that this would occur. The LPIS should not be considered a LSSS for protection of SL 2.1.1.1 since it does not provide a "significant safety function" with regard to protecting fuel cladding integrity. Critical power ratio continues to increase with decreasing reactor pressure (to at least pressures as low as 600 psia based on historical GE critical power testing), so there is not threat to fuel cladding integrity for this event. Even though there is no safety significance to this condition, it is a condition that may cause SL 2.1.1.1 to be exceeded.

Justification for Continued Operation

The purpose of reactor core SLs is to protect fuel cladding integrity such that significant fuel damage is not calculated to occur for any steady-state operation condition, normal operation transient, or AOO. The PRFO does not threaten fuel cladding integrity. Specifically, the Minimum Critical Power Ratio Safety Limit (SLMCPR) is defined to ensure fuel cladding integrity is protected. This event causes the Critical Power Ratio (CPR) to increase. With an initial condition that is restricted by the MCPR Operating Limit (OLMCPR) and an event that causes the CPR to increase, the margin to the SLMCPR increases during the event and therefore no threat to fuel cladding integrity exists.

Even if a PRFO were to occur in the interim before the overly conservative condition in SL 2.1.1.1 is addressed by a TS modification, SL 2.1.1.1 may not be exceeded due to the actual plant conditions when the transient occurred, including an actual LPIS value that may be significantly above the LPIS-AL used in the GE analysis.

An immediate plant change such as increasing the LPIS-AL is unfounded and could potentially degrade plant safety. Licensees have set the LPIS at the current plant values for a variety of reasons. Raising the LPIS-AL may cause unnecessary safety system challenges by increasing the incidence of plant isolations when, as described above, the PRFO event does not challenge fuel cladding integrity. Increasing the possibility of safety system challenges when not necessary could adversely affect plant safety and, therefore, is not appropriate.

Corrective/Preventive Actions

The GE evaluation has shown a vulnerability to exceeding a reactor core SL, for a condition that does not threaten fuel cladding integrity in any way. In this case, there is no clear compensatory action that can be defined to appropriately mitigate this vulnerability. Since the condition does not challenge the physical barrier that the SL is intended to protect, the LPIS should not be considered as a LSSS for SL 2.1.1.1 and there is no need for a compensatory action.

The BWROG Technical Specification Improvement Coordination Committee (TSICC) is evaluating this condition and the specification of SL 2.1.1.1. GE and the BWROG will develop a proposed resolution to this issue and interact with the NRC to obtain approval of the proposed TS modification. GE and the BWROG are willing to meet with the NRC at their convenience to discuss the situation and the potential approaches for a TS modification. Plant Technical Specifications and Bases will be developed as necessary, depending upon the outcome of the TS modification activity.

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Table 1. LPIS-AL* and MCFL Values Contained in GE Records

Utility	Plant	Cycle	LPIS-AL (PSIG)	MCFL (%)
AmerGen Energy Co.	Clinton	10	825	115
AmerGen Energy Co.	Oyster Creek	20	825	110
Carolina Power & Light Co.	Brunswick 1	15	785	115
Carolina Power & Light Co.	Brunswick 2	16	785	115
Constellation Nuclear	Nine Mile Point 1	19	834.2	110
Constellation Nuclear	Nine Mile Point 2	10	720	115
Detroit Edison Co.	Fermi 2	11	720	115
Energy Northwest	Columbia			
Entergy Nuclear Northeast	FitzPatrick	17	825	115
Entergy Nuclear Northeast	Pilgrim	16	782.3	109
Entergy Nuclear Northeast	Vermont Yankee	24	765	
Entergy Operations, Inc.	Grand Gulf	11	837	
Entergy Operations, Inc.	River Bend	10	825	
Exelon Generation Co.	Dresden 2	19	785	110
Exelon Generation Co.	Dresden 3	19	785	110
Exelon Generation Co.	LaSalle 1	11	825	120
Exelon Generation Co.	LaSalle 2	11	825	120
Exelon Generation Co.	Limerick 1	11	720	115
Exelon Generation Co.	Limerick 2	9	720	115
Exelon Generation Co.	Peach Bottom 2	16	750	115
Exelon Generation Co.	Peach Bottom 3	15	750	115
Exelon Generation Co.	Quad Cities 1	19	785	110
Exelon Generation Co.	Quad Cities 2	18	785	110
FirstEnergy Nuclear Operating Co.	Perry 1	11	782.3	123
Nebraska Public Power District	Cooper	23	825	110
Nuclear Management Co.	Duane Arnold	20	800	125
Nuclear Management Co.	Monticello	23	809	108
PPL Susquehanna LLC.	Susquehanna 1			
PPL Susquehanna LLC	Susquehanna 2			
PSEG Nuclear	Hope Creek	13	720	130
Southern Nuclear Operating Co.	Hatch 1	22	795	115
Southern Nuclear Operating Co.	Hatch 2	19	795	115
Tennessee Valley Authority	Browns Ferry 1	6	825	
Tennessee Valley Authority	Browns Ferry 2	13	825	
Tennessee Valley Authority	Browns Ferry 3	11	825	

*LPIS-AL values are turbine inlet pressure

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Please let me know if the NRC has any questions on this information. A verbal description was provided on this date to the NRC project manager for GE, and to representatives of the reactor system and technical specifications branches. I can be reached at (910) 675-6608, or at jason.post@ge.com.

Sincerely,

Jason Post

Jason. S. Post, Manager
Engineering Quality & Safety Evaluations

cc: S. D. Alexander (NRC-NRR/DISP/PSIB) Mail Stop 6 F2
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J. F. Klapproth (GE)
H. J. Neems (GE)
L. M. Quintana (GE)
T. Rumsey (GE)
PRC File

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Attachment 1 - Notified Plants

Affected	Potentially Affected	Utility	Plant
	X	AmerGen Energy Co.	Clinton
	X	AmerGen Energy Co.	Oyster Creek
	X	Carolina Power & Light Co.	Brunswick 1
	X	Carolina Power & Light Co.	Brunswick 2
	X	Constellation Nuclear	Nine Mile Point 1
X		Constellation Nuclear	Nine Mile Point 2
X		Detroit Edison Co.	Fermi 2
		Dominion Generation	Millstone 1
		Energy Northwest	Columbia
	X	Entergy Nuclear Northeast	FitzPatrick
X		Entergy Nuclear Northeast	Pilgrim
X		Entergy Nuclear Northeast	Vermont Yankee
	X	Entergy Operations, Inc.	Grand Gulf
	X	Entergy Operations, Inc.	River Bend
		Exelon Generation Co.	CRIT Facility
	X	Exelon Generation Co.	Dresden 2
	X	Exelon Generation Co.	Dresden 3
	X	Exelon Generation Co.	LaSalle 1
	X	Exelon Generation Co.	LaSalle 2
X		Exelon Generation Co.	Limerick 1
X		Exelon Generation Co.	Limerick 2
X		Exelon Generation Co.	Peach Bottom 2
X		Exelon Generation Co.	Peach Bottom 3
	X	Exelon Generation Co.	Quad Cities 1
	X	Exelon Generation Co.	Quad Cities 2
X		FirstEnergy Nuclear Operating Co.	Perry 1
	X	Nebraska Public Power District	Cooper
	X	Nuclear Management Co.	Duane Arnold
	X	Nuclear Management Co.	Monticello
		Pooled Equipment Inventory Co.	PIM
		PPL Susquehanna LLC	Susquehanna 1
		PPL Susquehanna LLC	Susquehanna 2
X		PSEG Nuclear	Hope Creek
	X	Southern Nuclear Operating Co.	Hatch 1
	X	Southern Nuclear Operating Co.	Hatch 2
	X	Tennessee Valley Authority	Browns Ferry 1*
	X	Tennessee Valley Authority	Browns Ferry 2
	X	Tennessee Valley Authority	Browns Ferry 3

*The plant is in an extended shutdown

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<u>Attachment 2</u> - Reportable Condition per §21.21(d)

(i) Name and address of the individual informing the Commission:

J. S. Post, Manager, Engineering Quality & Safety Evaluations, GE Energy - Nuclear, 3901 Castle Hayne Road, Wilmington, NC 28401.

(ii) Identification of the facility, the activity, or the basic component supplied for such facility or such activity within the United States which fails to comply or contains a defect:

The affected and potentially affected plants are identified in Attachment 1.

(iii) Identification of the firm constructing the facility or supplying the basic component which fails to comply or contains a defect:

GE Energy – Nuclear, Wilmington, NC.

(iv) Nature of the defect or failure to comply and safety hazard which is created or could be created by such defect or failure to comply:

The defect is a calculation of an anticipated operational occurrence (AOO), which predicts that the Pressure Regulator Failure Maximum Demand (Open) (PRFO) transient will be terminated by a high water level trip as a result of level swell in the reactor. An improved (and approved) model predicts that MSIV closure will occur when steam line pressure reaches the low-pressure isolation setpoint (LPIS), rather than terminate due to a high water level trip. Depending upon the plant-specific response to a PRFO, including the value of the LPIS, reactor steam dome pressure could decrease to below 785 psig while thermal power exceeds 25% of rated, which would be a violation of SL 2.1.1.1. This constitutes a *Defect* as defined in 10 CFR 21.3, even though there is no safety hazard created. SL 2.1.1.1 was intended to protect fuel cladding integrity during startup conditions without the need to perform a Critical Power Ratio (CPR) calculation. The AOO of concern is a transient from normal operating conditions that causes CPR to increase, so the event produces additional margin to the Minimum Critical Power Ratio Safety Limit (SLMCPR) and does not threaten fuel cladding integrity. The LPIS should not be considered as a Limiting Safety System Setting (LSSS) for SL 2.1.1.1 since it does not provide a "significant safety function" with regard to protecting fuel cladding integrity. This indicates that SL 2.1.1.1 is overly conservative because an event that causes CPR to increase and does not threaten fuel cladding integrity, may result in exceeding a reactor core SL.

(v) The date on which the information of such defect or failure to comply was obtained:

This was Identified as a potentially reportable condition by GE on January 28, 2005.

(vi) In the case of a basic component which contains a defect or failure to comply, the number and locations of all such components in use at, supplied for, or being supplied for one or more facilities or activities subject to the regulations in this part:

The basic component which contains the defect is a licensing analysis for a Pressure Regulator failure – Maximum Demand (Open) (PRFO) that shows reactor water level swell caused by the event will lead to turbine trip and the event will be terminated by the pressurization associated with the turbine trip. Updated GE models indicate that reactor pressure will decrease for this event until the MSIV low-pressure isolation setpoint (LPIS) is reached. GE analysis of the PRFO assuming MSIV trip is initiated at the LPIS-Analytical Limit, indicates that this can lead to exceeding Technical Specification SL 2.1.1.1.

(vii) The corrective action which has been, is being, or will be taken; the name of the individual or organization responsible for the action; and the length of time that has been or will be taken to complete the action (note, these are actions specifically associated with the identified Reportable Condition):

The GE evaluation has shown a vulnerability to exceeding a reactor core SL, for a condition that does not threaten fuel cladding integrity in any way. In this case, there is no clear compensatory action that can be defined to appropriately mitigate this vulnerability, and since the condition does not challenge the physical barrier that the SL intends to protect (i. e., the fuel cladding integrity), there is no safety basis for a compensatory action.

GE and the BWROG Technical Specification Improvement Coordination Committee (TSICC) is evaluating this condition and the specification of SL 2.1.1.1. GE and the BWROG are willing to meet with the NRC at their convenience to discuss the situation and the potential approaches for a TS modification. The BWROG will develop a proposed resolution to this issue and interact with the NRC to obtain approval of the proposed TS modification.

(viii) Any advice related to the defect or failure to comply about the facility, activity, or basic component that has been, is being, or will be given to purchasers or licensees:

The purpose of reactor core SLs is to protect fuel cladding integrity such that significant fuel damage is not calculated to occur for any steady-state operation condition, normal operation transient, or AOO. The PRFO does not threaten fuel cladding integrity. Specifically, the Minimum Critical Power Ratio Safety Limit (SLMCPR) is defined to ensure fuel cladding integrity is protected. This event causes the Critical Power Ratio (CPR) to increase. With an initial condition that is restricted by the MCPR Operating Limit (OLMCPR) and an event that causes the CPR to increase, there is increased margin to the SLMCPR during the event and there is no threat to fuel cladding integrity.

Even if a PRFO were to occur in the interim before the overly conservative condition in SL 2.1.1.1 is addressed by a TS modification, SL 2.1.1.1 may not be exceeded due to the actual plant conditions when the transient occurred, including an actual LPIS value that may be significantly above the LPIS-AL used in the GE analysis.

An immediate plant change such as increasing the LPIS-AL is unfounded and could potentially degrade plant safety. Licensees have set the LPIS at the current plant values for a variety of reasons. Raising the LPIS-AL may result in unnecessary safety system challenges by increasing the incidence of plant isolations when, as described above, this event does not challenge fuel cladding integrity. Increasing the possibility of safety system challenges when not necessary could adversely affect plant safety, and therefore, is not appropriate.