Pacific Gas and Electric Company

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December 2, 2003

PG&E Letter DCL-03-162

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

Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
Diablo Canyon Units 1 and 2
Response to NRC Request for Additional Information Regarding License Amendment Request 03-02, "Response Time Testing Elimination and Revision to Technical Specification 3.3.1, 'Reactor Trip System (RTS) Instrumentation'"

Dear Commissioners and Staff:

PG&E Letter DCL-03-016, dated February 28, 2003, submitted License Amendment Request (LAR) 03-02 which proposes to revise Technical Specification 3.3.1, "Reactor Trip System (RTS) Instrumentation," to add Surveillance Requirement 3.3.1.16 to function 3.a, Power Range Neutron Flux Rate - High Positive Rate Trip in Table 3.3.1-1. In addition, LAR 03-02 proposes to eliminate periodic pressure sensor response time testing (RTT) in accordance with WCAP-13632-P-A, Revision 2, "Elimination of Pressure Sensing Response Time Testing Requirements," and to eliminate periodic protection channel RTT in accordance with WCAP-14036-P-A, Revision 1, "Elimination of Periodic Protection Channel Response Time Tests."

On July 1 and July 25, 2003, the NRC staff identified additional information required to complete the evaluation associated with PG&E LAR 03-02. PG&E's response to the July 25, 2003 request for additional information (RAI) was provided in PG&E Letter DCL-03-137, dated October 30, 2003. PG&E's response to the July 1, 2003, RAI is included in Enclosure 1.

The additional information does not affect the results of the safety evaluation or no significant hazards consideration determination previously transmitted in PG&E Letter DCL-03-016.

A001

If you have any questions regarding this response, please contact Stan Ketelsen at 805-545-4720.

Sincerely,

David H. Oatley
Vice President and General Manager - Diablo Canyon

mjr/4557
Enclosures

cc:	Edgar Bailey, DHS
	Bruce S. Mallett
	David L. Proulx
	Diablo Distribution
cc/enc:	Girija S. Shukla

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

	)	Docket No. 50-275
In the Matter of	)	Facility Operating License
PACIFIC GAS AND ELECTRIC COMPANY	)	No. DPR-80
	)	
Diablo Canyon Power Plant	)	Docket No. 50-323
Units 1 and 2	)	Facility Operating License
	)	No. DPR-82

AFFIDAVIT

David H. Oatley, of lawful age, first being duly sworn upon oath says that he is Vice President and General Manager - Diablo Canyon of Pacific Gas and Electric Company; that he has executed this response to the request for additional information regarding License Amendment Request LAR 03-02 on behalf of said company with full power and authority to do so; that he is familiar with the content thereof; and that the facts stated therein are true and correct to the best of his knowledge, information, and belief.

David H. Oatley
Vice President and General Manager - Diablo Canyon

Subscribed and sworn to before me this 2nd day of December 2003.

Notary Public
County of San Luis Obispo
State of California

CHUCK MACKEY
Commission # 1397547
Notary Public - California
San Luis Obispo County
My Comm. Expires Feb 1, 2007

PG&E Response to NRC Request for Additional Information Regarding License Amendment Request 03-02, "Response Time Testing Elimination and Revision to Technical Specification 3.3.1, 'Reactor Trip System (RTS) Instrumentation'"

Questions received on July 1, 2003

NRC Question 1:

PG&E's amendment request stated that Westinghouse recently identified the need to credit the PFRT function to provide RCS overpressurization protection during a RWAP event. Specifically, the licensee stated that a generic Westinghouse evaluation showed that a low power RWAP event could result in the RCS pressure exceeding the 110 percent design limit (2750 psia) if only the typically credited trip functions (high pressurizer pressure, overtemperature delta-T, and power range neutron flux - high) are credited and assuming that the pressurizer pressure control system malfunctions. Since the licensee's amendment request proposes to make a change to the licensing basis of the DCPP accident analyses, specifically the RWAP analysis, the staff requests the licensee submit the new RWAP accident analysis for the staff's review. The licensee's analysis must, at a minimum, contain the following:

a. A description of the initial conditions and assumptions used in the analysis and a detailed justification explaining how each will ensure the maximum peak RCS pressure is calculated.

b. A description of the analytical methods and computer codes used to perform the analysis including references which state these methods and codes have previously been reviewed and approved by the staff.

c. A summary of the sequence of events including the time intervals between major events (i.e. reactor trip signals, major equipment operation, operational limits being reached, etc.).

d. A description of the single failure event assumed in the analysis, a comparison of how it affected the peak RCS pressure, and a technical justification for why it constitutes the worst-case single failure event.

e. A summary of the results obtained from the analysis including a comparison of these results to the DCPP TS safety limits to demonstrate acceptability.

PG&E Response 1.a:

The plant response during an uncontrolled rod control cluster assembly (RCCA) bank withdrawal at power (RWAP) event can vary significantly depending on the reactivity insertion rate, initial power level, and core reactivity feedback effects. The RWAP event results in an addition of positive reactivity into the core and an increase in the core heat flux. Since the heat extraction from the steam generator (SG) lags behind the core power generation until the SG pressure reaches the relief or safety valve setpoint, there is a net increase in the reactor coolant temperature. Unless terminated by manual or automatic action, the power mismatch and resultant coolant temperature rise could eventually result in a departure from nucleate boiling (DNB) or reactor coolant system (RCS) overpressure condition.

As discussed in the license amendment request (LAR) 03-02, the Diablo Canyon Power Plant (DCPP) Final Safety Analysis Report Update (FSARU) Section 15.2.2 presents the analysis of the RWAP event to verify that the reactor protection system is designed to terminate any such transient before the departure from nucleate boiling ratio (DNBR) falls below the safety analysis limit. This response describes the Westinghouse generic analysis of the RWAP overpressure event to verify that the reactor protection system is designed to terminate any such transient without resulting in the RCS pressure exceeding the maximum allowable 110 percent of the design value. The causes and description of the RWAP overpressure event are identical to those already discussed in FSARU Section 15.2.2 and the only significant difference from the DNBR analysis is that the generic RWAP overpressure analysis uses assumptions and initial conditions to make the event more limiting with respect to RCS overpressure. In order to obtain conservative RCS overpressure results, the following assumptions are made for the range of generic RWAP cases evaluated:

(1) The initial NSSS power is assumed to be 8 percent, which is the minimum power level at which the High flux reactor trip low setting (25 percent) can be blocked (10 percent) minus a 2 percent uncertainty. Starting at lower power allows the core power to increase and the maximum power mismatch to occur before a reactor protection setpoint is reached.

(2) Minimum reactivity feedback is assumed which allows the core power to increase more rapidly, and results in a greater power mismatch between the primary and secondary.

(3) The positive reactivity insertion rate is evaluated over the same conservatively bounding range of values as for the RWAP DNB analysis in the FSARU. The minimum reactivity insertion rate analyzed is 20 pcm/sec, since sensitivity studies indicated that RCS overpressure is not a concern for smaller insertion rates. The maximum reactivity insertion rate analyzed is 110 pcm/sec, which exceeds the maximum possible from the simultaneous withdrawal of the two control rod banks having the maximum combined worth at the maximum speed.

(4) The assumed RCS flow is based on a conservative thermal design flow rate.

(5) The RCS T_{avg} is assumed to be at the maximum programmed value for the initial power level plus a conservative uncertainty allowance, since the

rate of liquid expansion becomes more severe with increased temperature.

(6) The analysis assumes the maximum initial programmed value for the pressurizer water level including uncertainty. This minimizes the available pressurizer vapor volume space and maximizes the net pressurization effect for a given pressurizer liquid insurge.

(7) The effects of the initial RCS pressure were not unidirectional so cases were evaluated for both the maximum and minimum initial RCS pressure including uncertainty.

(8) There is no credit for the pressurizer power operated relief valves' (PORVs) relief capacity.

(9) There is no credit for the pressurizer spray system to control RCS pressure.

(10) There is no credit for the secondary steam dump control system.

(11) The pressurizer pressure safety valve (PSV) lift setpoints are assumed to be at their maximum values including a 3 percent setpoint tolerance plus a 1 percent setpoint shift and a maximum PSV loop seal purge delay time per WCAP-12910 (Reference 2).

(12) The analysis assumes a maximum pressurizer surge line friction factor which maximizes the pressure drop between the RCS and pressurizer. This maximizes the peak RCS pressure during the PSV relief conditions.

(13) The analysis assumes the Main Steam Safety Valves (MSSV) are at their maximum lift setpoints including a 3 percent setpoint tolerance and a 3 percent accumulation to full open.

(14) The reactor trip on high neutron flux is assumed to be actuated at a conservative value of 118 percent of nominal full power. The ΔT trips include all adverse instrumentation and setpoint errors, while the delays for the trip signal actuation are assumed at their maximum values.

(15) The reactor trip on high pressurizer pressure is assumed to be actuated at a conservative value of 2440 psia, which includes maximum uncertainty with a delay time of 2 seconds.

(16) For the cases which credit the positive flux rate trip (PFRT), the reactor trip is assumed to be actuated at a conservative setpoint value of 9 percent/second with a time constant of 2 seconds and a conservative bounding delay time of 3 seconds.

(17) The RCCA trip insertion characteristic is based on the assumption that the highest worth assembly is stuck in its fully withdrawn position.

PG&E Response 1.b:

The RWAP overpressure transient is analyzed by the LOFTRAN code, which is approved by the NRC for non-loss of coolant accident (LOCA) analysis in WCAP-7907-A, dated April 1984 (Reference 1). This code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level. The RWAP overpressure analysis is analyzed using the same methodology, which is documented in the DCPP FSARU Section 15.2.2. Since the generic RWAP overpressure analysis is evaluating peak RCS pressure as opposed to DNBR limit criteria, the initial conditions and assumptions are based on standard thermal design values including the worst case uncertainties. The acceptance criterion for the RWAP overpressure analysis is to maintain the peak RCS pressure less than 110 percent of the design limit of 2750 psia.

PG&E Response 1.c:

The magnitude of the RCS pressure increase resulting from the RWAP event is a function of the reactivity insertion rate, the initial power level, and the amount of reactivity feedback. The potential for RCS overpressure increases as the time between the reactivity insertion and a reactor trip increases due to the time lag associated with transfer of the increased heat generated in the core through the fuel and into the RCS coolant. For small positive reactivity insertion rates, the nuclear power and RCS temperature increase relatively slowly and in equilibrium such that this thermal lag effect on RCS overpressure is not a concern. For large reactivity insertion rates at the end of core life conditions, there is a significant reactivity feedback effect such that the nuclear power and RCS temperature still increase in relative equilibrium and RCS overpressure is not a concern. However, for large reactivity insertion rates at the beginning of core life with minimum reactivity feedback effects, the nuclear power increases much faster than the rate at which the energy can be transferred into the RCS. Sensitivity studies confirm that the limiting RWAP overpressure case initiates from a low power level, since this maximizes the net power mismatch between the reactor and the secondary before a reactor trip is actuated. The power mismatch that exists when the reactor trip occurs determines the magnitude of the insurge into the pressurizer and the RCS pressure overshoot that occurs above the PSV setpoint.

Figure 1 shows the trend of the peak RCS pressure results for the two limiting RCS overpressure cases versus the bounding range of reactivity insertion rates evaluated. Case 1 assumes a minimum initial RCS pressure (Po – uncertainty), while Case 2 assumes the maximum initial RCS pressure (Po + uncertainty). Both cases are first analyzed without credit for the PFRT function, and rely on either the high pressurizer pressure trip (HPPT) or the high neutron flux trip (HNFT). Figure 1 shows that as the reactivity insertion rate becomes smaller the RCS peak pressure becomes less limiting. For small reactivity insertion rates, the nuclear power and RCS temperature increase relatively slowly, such that the MSSVs are able to open and limit the power mismatch until the reactor trip is actuated. Figure 1 shows that for these small insertion rates the reactor trip occurs soon enough such that the PSV relief capability maintains the RCS within the 2750 psia limit.

The sequence of events and their associated times are very similar for the Case 1 and Case 2 results. The following Table 1 lists the Case 2 sequence of events for characteristic small, intermediate, and large reactivity insertion rate results, respectively, without any credit for the PFRT function. The Case 2 results show that for reactivity insertion rates less than about 90 pcm/sec, the reactor trip occurs on the HPPT signal, while for greater insertion rates, the trip occurs on the high neutron flux signal. For the Case 1 results this transition in which reactor trip signal actuates occurs at a slightly lower insertion rate of about 70 pcm/sec.

At greater reactivity insertion rates, the reactor power begins increasing faster than the primary system can transfer the heat to the secondary, and the power mismatch continues to increase until the reactor trip occurs and the control rods insert into the core. Figure 1 shows that for smaller to intermediate reactivity insertion rates, the Case 1 (Po – uncertainty) results are slightly more limiting than Case 2 (Po + uncertainty) results. The peak pressure results for both cases become more limiting as the reactivity insertion rate increases, since this results in a greater power mismatch at the time of the reactor trip. Figure 1 shows that when the reactivity insertion rate approaches a value in the range 30 to 50 pcm/sec, the resultant power mismatch at the time the reactor trip occurs could cause the RCS pressure to exceed the acceptable limit of 2750 psia.

Figure 2 shows the trend of the peak RCS pressures obtained for the Case 1 and Case 2 results over the same spectrum of reactivity insertion rates when the PFRT function is credited for mitigation. Table 2 lists the sequence of events for the same Case 2 characteristic small, intermediate, and large reactivity insertion rate results, with the PFRT function credited for mitigation. For the small reactivity insertion case, the power increases slowly enough such that the PFRT trip setpoint is not generated, and the Table 2 results are unchanged. Similar to Figure 1, Figure 2 shows that for these small reactivity insertion rates, the HPPT signal is actuated in time to maintain the power mismatch and the RCS pressure below the limit. As the reactivity insertion rate increases, the reactor power increase eventually becomes rapid enough such that the PFRT reactor trip setpoint is actuated. As the reactivity insertion rate becomes larger, the PFRT reactor trip continues to be generated earlier in the RWAP event. For large reactivity insertion rates, the nuclear power increase results in a faster PFRT reactor trip actuation, there is less net heat transferred to the RCS, and the RCS pressure response becomes less limiting. As Figure 2 shows, the conservative modeling of the PFRT trip provides adequate protection for a bounding range of reactivity insertion rates to ensure that a RWAP event does not result in the RCS pressure exceeding the limit.

PG&E Response 1.d:

The analysis assumes a most limiting single failure of one reactor protection train, such that the other train is still available for automatic actuation. Just as in the RWAP DNBR event in the DCPP FSARU, the RWAP overpressure event only requires an automatic reactor trip for mitigation, and there are no credible single failures of equipment which could make the event more severe.

PG&E Response 1.e:

Table 3 summarizes the key plant parameters assumed in the generic RWAP Overpressure analysis and compares them to the corresponding DCPP safety analysis parameters. The assumptions in the generic RWAP overpressure analysis are considered to be conservatively bounding for the comparable DCPP parameters. The DCPP low side uncertainty for pressurizer pressure is slightly greater than that assumed in the generic analysis. The HPPT setpoint is also slightly greater than in the generic analysis. However, these differences are minor and are more than offset by the large amount of conservatism that the generic analysis assumes in the other key parameters. As Table 3 shows, the generic analysis assumed greater values for the initial core power, initial RCS Tavg, the PFRT trip setpoint and delay, and the PSV setpoint tolerance and delay, which yields a significantly more conservative and bounding analysis. The Westinghouse generic RWAP overpressure analysis assumptions and results are applicable and conservatively bounding for DCPP.

In summary, the Westinghouse generic RWAP overpressure analysis confirms that the DCPP high pressurizer pressure and PFRT trip channels provide adequate protection over the entire range of possible reactivity insertion rates and plant conditions to ensure that the RCS pressure remains less than the safety analysis limit which is 110 percent of the design value.

Table 1: RWAP Overpressure Sequence of Events for Case 2 Results – No Credit for PFRT

Event Description	Small Reactivity Insertion Rate Case (20 pcm/sec)	Intermediate Reactivity Insertion Rate Case (60 pcm/sec)	Large Reactivity Insertion Rate Case (110 pcm/sec)
Initiation of uncontrolled RCCA withdrawal	0 sec	0 sec	0 sec
Reactor trip setpoint reached	20.2 sec (HPPT)	10.3 sec (HPPT)	7.7 sec (HNFT)
RCCAs begin to fall into core	22.2 sec	12.3 sec	8.2 sec
Peak RCS pressure occurs	2592 psia @ 24.9 sec	2830 psia @ 14.3 sec	3048 psia @ 10.3 sec

Table 2: RWAP Overpressure Sequence of Events for Case 2 Results – With Credit for PFRT

Event Description	Small Reactivity Insertion Rate Case (20 pcm/sec)	Intermediate Reactivity Insertion Rate Case (60 pcm/sec)	Large Reactivity Insertion Rate Case (110 pcm/sec)
Initiation of uncontrolled RCCA withdrawal	0 sec	0 sec	0 sec
Reactor trip setpoint reached	20.2 sec (HPPT)	7.1 sec (PFRT)	3.7 sec (PFRT)
RCCAs begin to fall into core	22.2 sec	10.1 sec	6.7 sec
Peak RCS pressure occurs	2592 psia @ 24.9 sec	2617 psia @ 13.1 sec	2708 psia @ 9.5 sec

Table 3: Comparison of Generic RWAP Overpressure Analysis Assumptions and DCPP Parameters

Parameter	Generic Analysis	DCPP Unit 2*
Core Power	3608 MWt	3411 MWt
Power Uncertainty	± 2%	± 2%
Nominal RCS Pressure	2250 psia	2250 psia
Pressure Uncertainty	±50 psia	+41.7, -51.7 psia
Nominal Full Power RCS Tavg	588.4 °F	577.7 °F
High Flux Trip Setpoint	118%	118%
High Flux Trip Delay	0.5 sec	0.5 sec
High Pzr Pressure Trip Setpoint	2440 psia	2460 psia
HPPT Delay	2.0 sec	2.0 sec
PFRT Setpoint/Rate Time Const.	9.0% / 2 sec	5.0 % / 2 sec
PFRT Delay	3 sec	0.5 sec
PSV Setpoint	2500 psia	2500 psia
PSV Setpoint Tolerance (includes setpoint shift)	+4%	+2%
PSV Loop Seal Purge Delay	1.5 sec	1.27 sec

*DCPP Unit 2 is more limiting than Unit 1

Figure 1: Westinghouse Generic RWAP Overpressure Analysis
PWR without PFRT

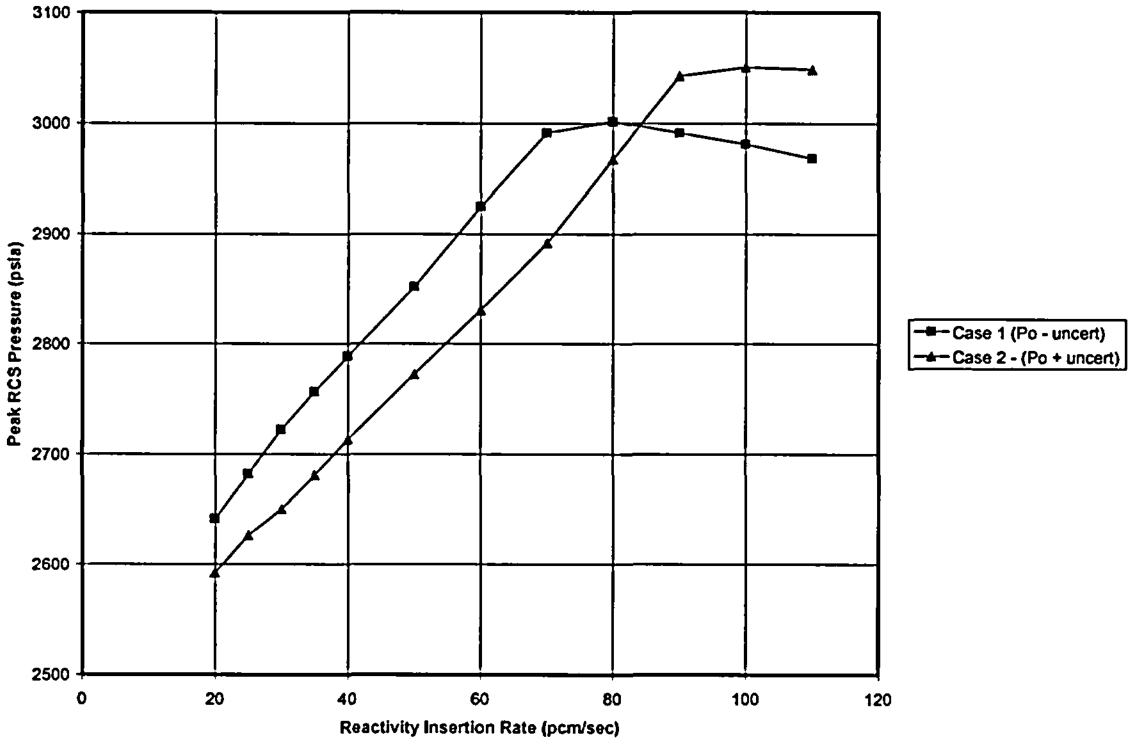
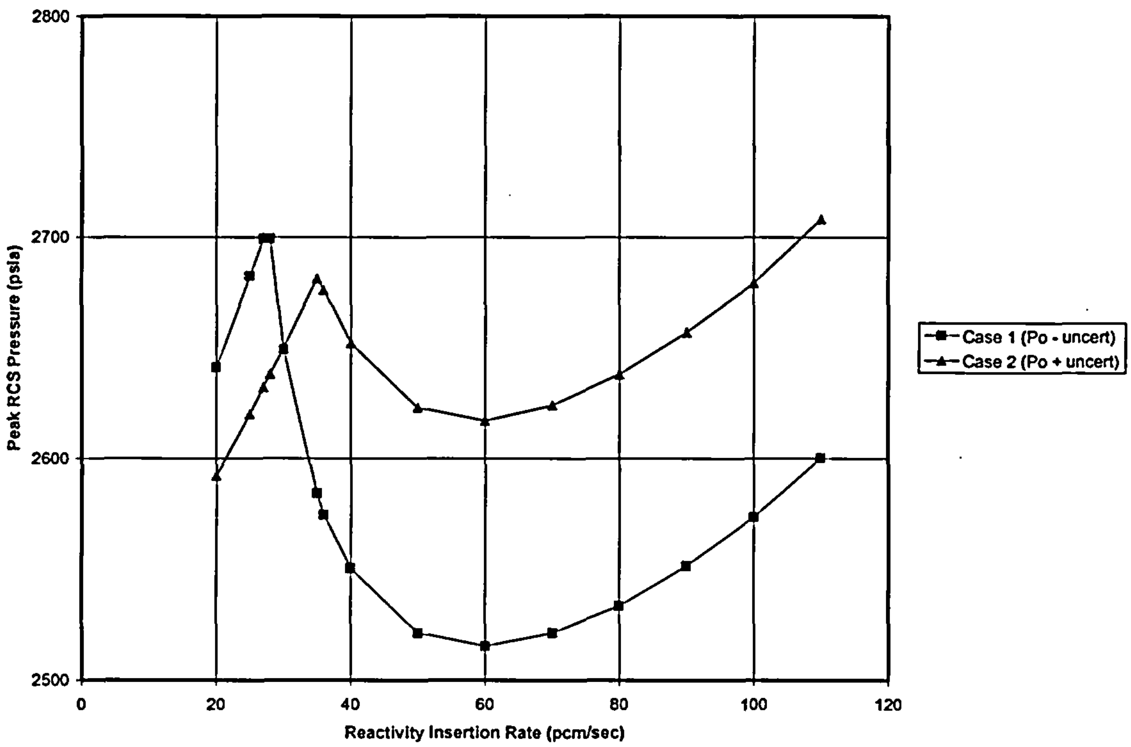


Figure 2: Westinghouse Generic RWAP Overpressure Analysis PWR with PFRT



NRC Question 2:

The licensee's amendment request indicates that DCPP had not previously identified the RWAP accident of low worth from middle to low power conditions as challenging the RCS overpressure Safety Limits. Since the new accident conditions were not previously considered as a design basis accident, it is likely that training programs and procedures do not adequately consider these conditions. The staff requests the licensee provide a summary of the changes it will make to the DCPP training program and procedures to ensure that operators and other plant personnel will be properly trained to take appropriate actions to mitigate the consequences of this new variant of the RWAP accident.

PG&E Response:

As discussed in Response 1, the only differences between the RWAP DNBR analyses currently documented in the FSARU Section 15.2 and the RWAP overpressure events discussed in this LAR 03-02 are the assumptions related to the initial plant conditions and control system performance. The current RWAP analyses in the DCPP FSARU assume nominal plant conditions and operation of the pressurizer pressure control system. The RWAP overpressure analyses assume the maximum uncertainty on the initial plant conditions and no operation of the pressurizer pressure control system. Westinghouse determined that the majority of the RWAP cases analyzed in the FSARU are adequately protected for both minimum DNBR and RCS overpressure concerns, based on the current response time assumptions for the HNFT and HPPT functions. As discussed in Response 1, only those RWAP events which occur from a low power level (10 percent) at the beginning of core life (minimum reactivity feedback), and with a medium to high reactivity insertion rate credit the PFRT function for mitigation.

There is no significant difference in the sequence of events, plant response, or timing of reactor protection actuation between the RWAP DNBR cases in the current FSARU and the RWAP overpessure events presented in this LAR. The RWAP overpressure cases and the comparable RWAP DNBR cases, all involve a positive reactivity insertion, which results in a prompt reactor trip due to an increase in core power, RCS temperature, and/or RCS pressure. The RWAP events are of short duration and only require actuation of a reactor trip signal for termination, such that operator involvement is insignificant. In summary, there are no new significant characteristics of the RWAP overpressure analysis compared to the current RWAP cases already analyzed in the FSARU. The only operator training required to implement this license amendment, with respect to the PFRT function, is to provide an update to the operators that response time testing of the PFRT function has been added to the technical specifications (TS) based on the Westinghouse generic safety analysis which credits this function for a limited spectrum of RWAP overpressure cases. As part of this training, the bases for this TS change will also be provided.

NRC Question 3:

The licensee's amendment request states that the Pressurizer High Water Level Trip is credited with preventing the pressurizer from becoming water solid during low worth and low power rod withdrawal accidents. However, the licensee also stated that the PFRT is credited during these same accidents with preventing the RCS overpressure condition. The staff requests the licensee clarify the need to credit two independent reactor trips during the same event. The occurrence of either trip should prevent both the overpressure and water solid conditions.

PG&E Response:

The Pressurizer High Water Level Trip (PHWLT) is not credited for protection for the same cases or criterion for which PFRT is credited. As noted in Response 1.c above, RCS overpressure may be a concern for rod withdrawal cases from lower power levels but with higher worth (i.e., higher reactivity insertion rates). The PHWLT is implicitly credited with preventing pressurizer overfill for lower worth rod withdrawals from low power levels, for which a PFRT, HNFT, or OTDT may not be actuated prior to overfill. However, the PHWLT is not credited for protection against DNB or RCS overpressure since normal operation of the pressurizer level control system (charging and letdown) may prevent the water level from increasing appreciably. This function is only implicitly credited with performing its intended function of preventing pressurizer overfill for RWAP cases where that criterion may be challenged. A further discussion of the PHWLT function is contained in NSAL-02-11 (Enclosure 2).

NRC Question 4:

The staff requests the licensee provide a copy of generic Westinghouse analysis (Reference 10 of licensee's amendment request) that served as the basis for amending the DCPP licensing basis.

PG&E Response:

Reference 10 is the Westinghouse Nuclear Safety Advisory Letter NSAL-02-11, which only provides a summary of the analysis and conclusions related to the RWAP Overpressure analyses. A copy of NSAL-02-11 is provided in Enclosure 2. PG&E has independently reviewed and verified the Westinghouse generic RWAP overpressure results and conclusions. Westinghouse performed the generic RWAP overpressure analysis using the same codes and methodology as currently documented in the DCPP FSARU.

References

1. LOFTRAN Code Description, WCAP-7907-A, T. W. T. Burnett, April 1984

2. Pressurizer Safety Valve Set Pressure Shift, WOG Project MUHP 2351/2352, WCAP-12910 Revision 1-A, G.O. Barrett, June 1993

Westinghouse Letter NSAL-02-11

Nuclear Safety Advisory Letter

Westinghouse Electric Company

This is a notification of a recently identified potential safety issue pertaining to basic components supplied by Westinghouse. This information is being provided so that you can conduct a review of this issue to determine if any action is required.

P.O. Box 355, Pittsburgh, PA 15230

Subject: **Reactor Protection System Response Time Requirements**	Number: **NSAL-02-11**
Basic Component: Reactor Protection System	Date: 07/29/2002
Plants: All Westinghouse NSSS	
Substantial Safety Hazard or Failure to Comply Pursuant to 10 CFR 21.21(a)	Yes ☐ No ☒
Transfer of Information Pursuant to 10 CFR 21.21(b)	Yes ☐
Advisory Information Pursuant to 10 CFR 21.21(d)(2)	Yes ☐
References: See attached	

SUMMARY

The Technical Specifications (Tech Specs) for most plants require response time testing of the specific Reactor Protection System (RPS) functions. The response time acceptance criteria are typically identified in a separate licensee-controlled document or in the Tech Specs (for plants that have not relocated them out of the Tech Specs). The response time specified for some functions may be "Not Applicable." Some of these protection functions may be credited for primary protection against anticipated transients or postulated accidents, but not explicitly in the specific safety analysis cases presented in the Final Safety Analysis Report (FSAR).

Recently, some licensees have determined that crediting a function that has no response time requirement for primary protection may impact function operability. The purpose of this communication is to identify protection system functions potentially in this category, explain the basis for the response time requirement designation of "Not Applicable," assess the safety significance of the issue, and identify Westinghouse conclusions.

Additional information, if required, may be obtained from the originators. Telephone 412-374-5773 or 412-374-5424.

Originator(s):

G. H. Heberle
Transient Analysis

D. S. Huegel
Transient Analysis

Approved:

H. A. Sepp, Manager
Regulatory & Licensing Engineering

ISSUE DESCRIPTION

The Improved Standard Technical Specifications (ISTS) for Westinghouse plants, NUREG-1431 (Reference 1) includes requirements for the Reactor Trip System (RTS) and Engineered Safety Features Actuation System (ESFAS) instrumentation. The Tech Specs identify the applicable plant operational mode(s) and other specified conditions, the minimum number of channels that must be operable, and the nominal setpoint and/or allowable value for each function. In addition, there are surveillance requirements to verify that the RTS and ESFAS response times are within the values assumed in the safety analyses. The Standard Technical Specifications (STS) for Westinghouse plants, NUREG-0452 (Reference 2) contained specific response time values for each function. Plants that have implemented the ISTS (Reference 1) or relocated the response time tables out of the Tech Specs have the response time values in a separate licensee-controlled document, such as a Technical Requirements Manual, or in Chapter 7 or 16 of the Final Safety Analysis Report (FSAR). The response time requirement specified for some RTS and ESFAS functions is "Not Applicable" (or "N.A."). It should also be noted that for some older Westinghouse plants whose original Tech Specs pre-date the STS there are no requirements in the Tech Specs to perform RTS or ESFAS response time testing.

Recently, a "condition report" was prepared by a licensee when it was identified that Westinghouse had credited the Power Range Neutron Flux – High Positive Rate reactor trip function, commonly called the positive flux rate trip (PFRT), in a safety analysis. The discussion of this trip function in the technical evaluation section below provides more details on the analytical basis. The licensee concluded that it is necessary to verify the response time of the function since it is explicitly credited for primary protection. However, the Tech Spec response time requirement for the PFRT is listed as "Not Applicable," and this function had not been response time tested.

This communication discusses the basis for the response time requirements as originally presented in the STS, including the "Not Applicable" designation. In addition, other trip functions that may be similarly credited or recognized as providing a primary protective function are identified.

TECHNICAL EVALUATION

As noted in the STS (Reference 2), Bases Sections 3/4.3.1 and 3/4.3.2, Reactor Trip and Engineered Safety Feature Actuation System Instrumentation, "*No credit was taken in the analyses for those channels with response times indicated as not applicable.*" In practice, "*the analyses*" mentioned here referred to the specific safety analyses as presented in the FSAR accident analysis section (i.e., Chapter 14 or 15). Historically, response time values were defined in the STS for those RTS or ESFAS functions that were explicitly modeled and credited for primary protection in the FSAR analyses. The response time values for other protection functions not explicitly credited in these analyses were generally listed as "Not Applicable." This practice represents the original licensing basis approach for most plants. Plants that have relocated the response time values to licensee-controlled documents have typically retained the "Not Applicable" designations that were previously contained in their Tech Specs.

The historical designation of "Not Applicable" for a response time criterion should not be construed to imply that a particular RTS or ESFAS function is unimportant, or that it only provides backup protection. In fact, while not credited in the specific limiting analysis case(s) presented in the FSAR, some of these functions are relied upon to provide primary protection for a plant operational mode or condition that is not explicitly analyzed and/or presented. In general, all RTS and ESFAS protection system functions are important and required to be operable to ensure that the safety analysis basis remains valid and bounding and all design criteria (e.g., diversity, defense-in-depth) are satisfied.

Westinghouse has reviewed the generic response time requirements originally listed in the STS (Reference 2) for the RTS (Table 3.3-2) and ESFAS (Table 3.3-5), as well as a sampling of current plant-specific time response requirements. As a result, the following sections numbered 1 through 5 identify trip functions that are typically credited as providing primary protection, but are not necessarily explicitly modeled in the analyses presented in the FSAR. Consistent with the historical approach for specifying response times identified above, the response time requirement may be specified as "Not Applicable" for some of these functions contained in the licensee-controlled document or the Tech Specs. This list captures the functions Westinghouse has identified as potentially affected by this issue. However, protection system designs vary and not all plant-specific requirements are the same. Also, Westinghouse does not retain the safety analyses of record for some plants.

1. Power Range Neutron Flux – High Positive Rate Reactor Trip

The Uncontrolled Rod Cluster Control Assembly (RCCA) Bank Withdrawal at Power (RWAP) accident is analyzed in the FSAR to demonstrate that the departure from nucleate boiling (DNB) design basis is met. Therefore, the analysis assumptions are such that the DNB ratio is minimized, including the assumption that RCS pressure control systems (pressurizer spray and relief valves) are operable.

While analyzing a specific plant in the early 1990s, Westinghouse identified the potential for Reactor Coolant System (RCS) overpressurization for some cases of RWAP. It was found that some RWAP cases from a low power level, i.e., 10% Rated Thermal Power (RTP), may approach or exceed the applicable RCS pressure limit (110% of design pressure), given the typical conservative analysis methodology and assumptions. This could occur if the RCS pressure control systems are not operable and only the Power Range Neutron Flux - High Setpoint, Pressurizer Pressure - High, and Overtemperature ΔT reactor trip functions are credited. It was demonstrated that crediting the PFRT provided the necessary protection to prevent RCS overpressurization.

Subsequently, a generic analysis was performed to address this for other Westinghouse plants with the PFRT function for which Westinghouse maintains the safety analyses. As was done in the earlier plant-specific analysis, the RWAP RCS overpressure analysis employed conservative assumptions, including a conservative setpoint and a very long delay time for the PFRT. The nuclear instrumentation system (NIS) trip functions credited in the safety analyses typically assume a maximum delay time of 0.5 second. The RWAP RCS overpressure analysis conservatively assumed a much longer delay time of 3.0 seconds for the PFRT. Based on this, it was determined that the delay time of the PFRT function is not a critical parameter, and that specific response time testing to verify this assumption is not necessary. It was concluded that if the trip function is operable, as is required by the Technical Specifications, then it could reasonably be credited to actuate within the conservative delay time assumed in the RWAP RCS overpressure analysis. The conclusion was that plants with a PFRT function are adequately protected against RCS overpressurization from a RWAP event, and that plant-specific analyses are not required, assuming that the generic analysis assumptions are bounding for the plant. Thus, it was determined that the FSAR documentation of the plant-specific RWAP analysis, which focuses on the DNB criterion, remains adequate and appropriate.

Note that some older Westinghouse plants do not have a PFRT function. Specific RWAP RCS overpressure analyses have been performed for these plants for which Westinghouse maintains the safety analyses. For some of these plants it has been necessary to impose a reduced maximum allowable reactivity insertion rate in the reload core safety evaluation.

In addition to RWAP RCS overpressurization discussed above, it should also be noted that the PFRT function has always been recognized as providing a primary protective function for RCCA Ejection accidents. The PFRT complements the Power Range Neutron Flux - High and Low Setpoint functions that are credited in the zero power and full power analyses presented in the FSAR. Specifically, the PFRT provides a reactor trip in the event of a low-worth RCCA ejection from a part-power initial condition, for which the Power Range Neutron Flux - High setpoint may not be reached. This is the reason for its inclusion in the Reactor Protection System (RPS), as noted in the ISTS Bases description of the PFRT function (Reference 1). This has also been discussed in some plant-specific RPS design-basis documents provided by Westinghouse. The nominal setpoint of +5% RTP with a 2-second rate time constant was chosen generically based on scoping analyses to provide a desired trip in the event of a rapid power change indicative of a RCCA ejection, but typically avoid an unnecessary trip in the event of a load increase or load rejection transient. However, there are no generic or plant-specific RCCA ejection safety analyses that explicitly model the PFRT function or response time. While the trip response time would be expected to be consistent with the other power range NIS trip functions, it is not considered a critical parameter for a low-worth RCCA ejection from a part-power initial condition. If the trip function is operable, as required by the Tech Specs, then it can reasonably be assumed to actuate and provide the necessary protection.

Note that the PFRT was added on a forward-fit basis to all Westinghouse plant designs with a negative flux rate trip (NFRT) function. The PFRT was not backfitted to older Westinghouse plants (i.e., those without the NFRT) by virtue of their being licensed prior to the inclusion of this function in the protection system design and the very low probability of a part-power RCCA Ejection accident.

2. Source Range Neutron Flux Reactor Trip

The Source Range reactor trip function is not explicitly credited in the limiting RCCA Bank Withdrawal from Subcritical (RWFS) and RCCA Ejection accident analysis cases presented in the FSAR. As a result, the response time requirement for this function was noted as "Not Applicable" in the STS, consistent with the historical approach described above. Nevertheless, this trip function is recognized as providing primary protection against reactivity insertion events such as RWFS and RCCA Ejection from the lower plant operational modes, where the Power Range NIS trips are not required to be operable. Refer to NSAL-00-016 (Reference 3) for a more detailed discussion.

3. Overpower ΔT Reactor Trip

The Overpower ΔT reactor trip function was not explicitly credited in any of the specific safety analysis cases originally presented in the FSAR. As a result, the response time requirement for this function was noted as "Not Applicable" in the STS, consistent with the convention described above. However, as described in Reference 4, this trip function is designed to provide primary protection against fuel centerline melt as a result of excessive linear heat generation during postulated transients. It also limits the range over which the Overtemperature ΔT trip function is required to provide protection against DNB.

In addition, the Overpower ΔT trip function provides primary protection for certain main steam line break cases. The limiting main steam line break accident analysis traditionally presented in the FSAR is the core response from a zero power initial condition with control rods inserted in the core. This analysis bounds the post-trip phase of a steam line break occurring from an at-power initial condition. However, as described in the Westinghouse steam line break topical (Reference 5, Section 3.2), the Overpower ΔT reactor trip function provides primary protection against DNB for some intermediate break sizes from an at-power initial condition. The limiting full power case that trips on Overpower ΔT is normally the largest break size that is too small to generate a trip on the safety injection actuation from a steam line break protection function (e.g., Steam Line Pressure - Low).

Subsequent to the original FSAR, Westinghouse has performed specific full power steam line break analyses for many plants to demonstrate that the DNB design basis is met. Traditionally these analyses have not been explicitly documented in the FSAR. However, in recent years some plants have included documentation of the full power steam line break analysis in the FSAR.

Note that although the STS Revision 4 (Reference 2) specified "Not Applicable" for the response time of the Overpower ΔT trip function, the draft STS Revision 5, upon which some plant-specific Tech Specs were based, included a response time test criterion and surveillance requirement. A sampling of several plant-specific requirements shows that most now include a response time for this trip function.

4. Pressurizer Water Level – High Reactor Trip

As noted in the discussion on the PFRT function above, the RWAP analysis presented in the FSAR focuses on the primary concern for this event by demonstrating that the DNB design basis is met. However, another criterion for any incident of moderate frequency (ANS Condition II event) is that a more serious plant condition should not be generated without other faults occurring independently. For the RWAP event prior to reactor trip the pressurizer water level increases due to coolant expansion from the RCS heatup. This may lead to filling the pressurizer, which could in turn result in a discharge of water through the pressurizer safety valves, potentially damaging the valves such that the RCS pressure boundary cannot be subsequently isolated. Thus, a more serious plant condition could be created. These concerns are avoided for a RWAP by the presence of the Pressurizer Water Level – High function, which will trip the reactor prior to pressurizer overfill and potential water relief through the safety valves, as noted in the ISTS Bases description of this trip function (Reference 1). After the reactor trip the coolant in the RCS will contract and the pressurizer level will drop. This function is not explicitly credited in the FSAR analysis cases for DNB since the normal operation of the pressurizer level control system (charging and letdown) may prevent the level from increasing. However, the trip provides a primary protective function in the event that water level does increase and challenge the overfill criterion. The specific response time for this trip is not a critical parameter, since any trip that occurs in a reasonable time (i.e., within several seconds) after reaching the high level setpoint will prevent overfill. Thus, Westinghouse concluded that there is no need to impose a specific response time requirement on this basis. If the trip function is operable, as is required by the Tech Specs, then it can reasonably be assumed to actuate and provide the necessary protection.

5. Safety Injection Signal Actuation of Auxiliary Feedwater

The STS included response time requirements for the Auxiliary Feedwater Pump start via the specific ESFAS functions that result in actuation of a Safety Injection (SI) signal (see Reference 2, Table 3.3-5). This is consistent with the fact that the Small Break Loss of Coolant Accident (SBLOCA) analysis presented in the FSAR typically credits the Auxiliary Feedwater (AFW) start on the SI signal from Pressurizer Pressure - Low. In addition, steam line break mass and energy release analyses, used as input to environmental qualification of equipment in compartments outside containment, credit the actuation of AFW on the SI signal generated by the Steam Line Pressure - Low (or equivalent) function. For steam line break the actuation of AFW affects the time at which the steam generator tube bundle uncovery occurs, following which superheated steam is released out the break.

A review of some plant-specific requirements documents shows that the Auxiliary (or Emergency) Feedwater actuation response time for the specific ESFAS functions is identified as "Not Applicable." However, it appears that these function-specific items have been superseded by a separate system-level item (not found in the STS) which identifies a response time requirement for motor-driven AFW pump start from any SI signal.

SAFETY SIGNIFICANCE

The convention of specifying "Not Applicable" as a response time requirement for protection system functions that are not explicitly credited in the specific FSAR analysis cases was incorporated into the STS (Reference 2). However, some of these functions are relied upon either explicitly or implicitly to provide primary protection for other plant conditions, or to address other criteria that are secondary to those addressed in the FSAR. The Tech Specs require that the trip functions be operable in the applicable plant operational modes or other specified conditions. RTS and ESFAS channel operability is verified by performing Channel Operational Tests (COTs) and Channel Calibrations required by the Tech Specs. Any significant degradation in the response time of a trip channel is likely to be detected during these tests. Thus, despite the lack of an explicit response time criterion and associated response time testing, it is reasonable to conclude that the channels are operable, i.e., capable of performing their intended safety function, and will trip or actuate in a timely manner. This conclusion is supported by the fact that response time requirements are defined and routinely confirmed for other similar functions.

It should also be noted that Westinghouse provided startup test procedures to each plant, which typically included response time criteria for all of the protection functions, based upon the typical safety analysis assumptions or original equipment specifications for the trip function. Thus, some plants may have confirmed response times for all of these functions at least once during initial plant startup testing. Based on the foregoing, Westinghouse concludes that this does not constitute a substantial safety hazard or failure to comply pursuant to 10 CFR Part 21.

RECOMMENDED ACTIONS

The practice of not explicitly defining a response time criterion for functions that are not explicitly credited in the FSAR, but that are nevertheless relied upon for primary protection is based on historical precedent and the contents of the original FSAR. This practice is consistent with the original licensing basis for Westinghouse plants. As noted in the assessment of safety significance above, Westinghouse does not consider this to be a safety concern, assuming that plants perform the required Tech Spec surveillance testing to demonstrate that the RTS and ESFAS channels are operable. Consistent with the original licensing basis, Westinghouse concludes that changes to define specific requirements for RTS and ESFAS functions currently identified as "Not Applicable" are not required unless the safety analyses explicitly crediting these functions are documented in the FSAR.

REFERENCES

1. NUREG-1431, Standard Technical Specifications Westinghouse Plants, Revision 2, April 2001.
2. NUREG-0452, Standard Technical Specifications for Westinghouse Pressurized Water Reactors, Revision 4, Issued Fall 1981.
3. Westinghouse Nuclear Safety Advisory Letter, NSAL-00-016, Rod Withdrawal from Subcritical Protection in Lower Modes, December 4, 2000.
4. WCAP-8746-A, Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions, September 1986.
5. WCAP-9227-A, Revision 1, Reactor Core Response to Excessive Secondary Steam Releases, February 1998.