WESTINGHOUSE CLASS 3

ADDENDUM 1
WCAP-11326

WESTINGHOUSE OWNERS' GROUP
TRIP REDUCTION & ASSESSMENT PROGRAM

STEAM GENERATOR LOW WATER LEVEL PROTECTION SYSTEM MODIFICATIONS TO REDUCE FEEDWATER-RELATED TRIPS NRC REVIEW QUESTIONS AND RESPONSES

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The following are responses to NRC Staff review questions concerning the delayed steam generator low-low water level trip function design described in WCAP-11325 (P - proprietary) and WCAP-11326 (NP - non-proprietary), "Steam Generator Low Water level Protection System Modifications to Reduce Feedwater-Related Trips". These reports were submitted by the Westinghouse Owners Group for NRC review and approval in December, 1986. Questions were received in March, 1987. Draft responses were discussed at a meeting held on April 9th (minutes are available in Westinghouse Owners Group letter no. OG-87-25, May 11, 1987). Responses, which incorporate the results of this meeting, and of subsequent discussions, are hereby formally submitted as addenda to WCAP-11325 (P) and WCAP-11326 (NP).

Steam Generator Low Water Level Protection System
Modifications to Reduce Feedwater-Related Trips

Question:

1. Page 55, section 4.4 and page 69, second paragraph: These paragraphs present the explanation of the logic in a confusing way. It is noted that in both the "1 SG" and the "2 SG" trip actuation circuits the timers are downstream of "AND" gates, but the characteristic that distinguishes the two actuation circuits is that in the case of "1 SG", the timer is started by a signal from a single steam generator to an "OR" gate, whereas, in the "2 SG" case the timer is started by a signal from multiple steam generators to 2/4 logic. The paragraphs should be rewritten to clarify this point.

Response:

Sentence 2 of section 4.4 will be changed to the following:

> These transients would satisfy the AND gate that receives the trip signals from all steam generators.

Sentence 2, paragraph 2, page 55 will be changed to the following:

> These transients would satisfy the OR gate that receives the trip signals from all steam generators.

Sentences 2 & 3, paragraph 2, page 69 will be changed to:

> Allowable time delays traced by the "1 SG" curve would be set on the timers downstream of the OR gates which receive trip signals from all steam generators (see Figures 24 and 25). The "2 SGs"

curve would apply to timers downstream of the AND gates which receive trip signals from all steam generators.

Question:

2. This question has been deleted, as the result of discussion and agreement at the meeting of April 9th (see the meeting minutes: Westinghouse Owners Group letter OG-87-25 dated 5/11/87).

Question:

3. Page I-21, section 17.3: This section states that the auxiliary feedwater system has an inherent one minute delay before flow to the steam generators is established due to pump start up, valve opening times and transport lag. How has this one minute delay been factored into the delay timing curves in Figures 27, 28 and 29?

Response:

The time delays given in Figures 27, 28 and 29 are actually delays in the transmission of the low-low steam generator water level trip signal to the actuated devices. Any inherent delays, to account for pump start up, valve opening, and other mechanical, electrical, and fluid effects, which typically amount to one minute, are additional to the time delays of Figures 27, 28, and 29, and have been considered in the supporting accident analyses of WCAP-11325 (P) and WCAP-11326 (NP).

Question:

4. Pages 48 and 49, Figures 18 and 19: It is noted that a .22 sq. ft. break at 50% power appears to result in more severe consequences than a .89 sq. ft. break at 50% power (Figures 16 and 17). Therefore, the curves for a .22 sq. ft. break at 10% power should be given to show that this does not represent a more severe case than at 50% power. Westinghouse should then show that the introduction of a trip delay will not cause a challenge to pressurizer safety valves which would have been avoided if no time delay was introduced.

Response:

The feedwater line rupture analyses were performed over a range of power levels and break sizes to verify that the introduction of a delay to the low-low steam generator water level trip signal would continue to produce acceptable consequences for the feedwater line rupture accident (i.e., satisfy Condition IV acceptance criteria: maintenance of a coolable core geometry and compliance with 10CFR100 dose guidelines).

Since the cases analyzed were for verification purposes only, some very conservative assumptions were employed. Conservatism is introduced [

]+a,c Therefore, the feedwater line rupture transients of WCAP-11325 (P) and WCAP-11326 (NP), are basically a feasibility assessment, intended only to check the trip delay times. Analyses of these cases, assuming [ ]+a,c including the trip delays when applicable, would be expected to produce less severe consequences.

The transient curves for a 0.22 sq. ft. feedwater line rupture, occurring at 10 percent power, are provided in Figures 1, 2, and 3. The acceptance criteria for this accident are satisfied, despite some overly conservative assumptions. Application of more reasonable; but still conservative assumptions, would result in a less severe transient.

The question of increased challenges to the pressurizer safety valves is most important in the context of a relatively frequent event (e.g., Condition II, loss of normal feedwater accident). For the purpose of judging a possible increase in the liklihood of opening the pressurizer PORVs or safety valves during a loss of feedwater, it is appropriate to consider the normal operation of control systems, such as automatic rod control, pressurizer spray, and steam dumping through the steam generator PORVs. The loss of feedwater accident analyses of WCAP-11325 (P) and WCAP-11326 (NP) were performed assuming that the control systems were not available (for conservatism). In fact, the steam generator power-operated relief valves can be expected to open during a loss of feedwater accident. The steam generator PORVs can relieve approximately one-quarter of the nominal steam flow. This additional cooling would be effective in reducing the RCS pressurization and preventing the pressurizer relief and safety valves from opening, especially at the lower power levels, where the longer time delays would be applied.

For a two-timer system, which is expected to be the most popular option, trip delays would be applied only at power levels below P-7 (approximately 10% power) for low-low water level in more than one steam generator, and below P-8 (typically less than 50% power) for low-low water level in one steam generator. The four-timer system would impose very short time delays above these interlocks, since the time delays would be set for full power operation.

The results of sensitivity studies, with the steam generator PORVs assumed inoperative, indicate that the loss of feedwater to a single steam generator would [

]+a,c Loss of feedwater to all steam generators, [

]+a,c The trip delay, therefore, would not increase the liklihood that the pressurizer relief or safety valves would open.

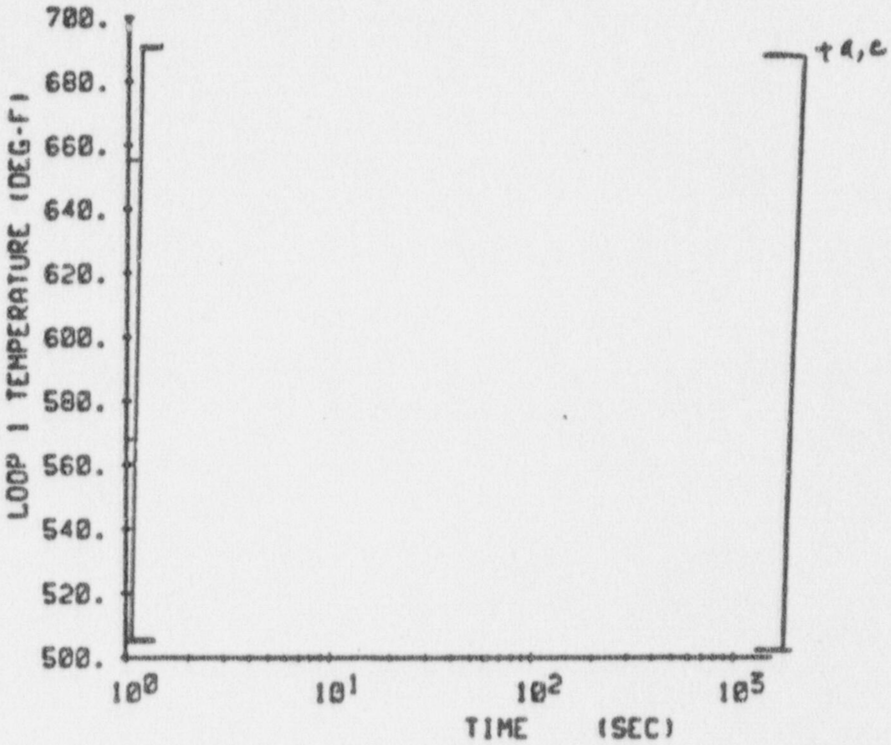

Figure 1

Faulted Loop Temperatures

Saturation, Hot Leg and Cold leg (4-loop plant)

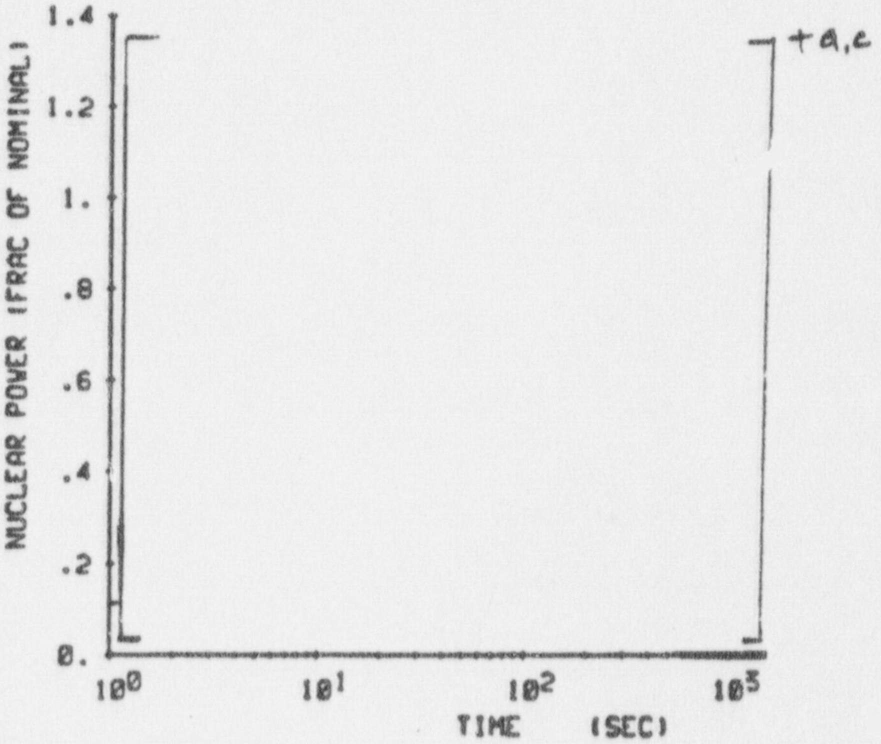

Figure 2
Nuclear Power (4-loop plant)

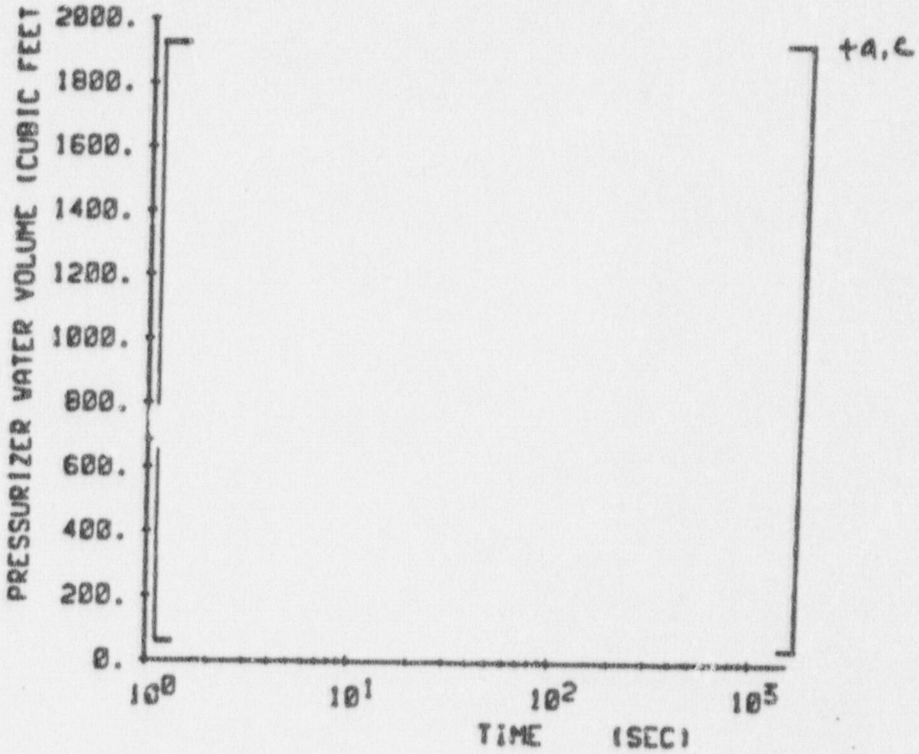

Figure 3
Pressurizer Water Volume (4-loop plant)

Steam Generator Low Water Level Protection System
Modifications to Reduce Feedwater-Related Trips

Question:

5. WCAP-11325 (P) and WCAP-11326 (NP) point out that some core designs have slightly positive moderator temperature coefficients at low power and/or early in core life. For an accident that causes low-low steam generator level and increased core temperature, the reactor trip is caused by exceeding the P-7 interlock setpoint. This causes a delay in trip initiation until the P-7 interlock is exceeded. Provide the results of the analyses that have been conducted to determine the effects of this trip delay and power excursion on the core? Will plant-specific studies be conducted in this area?

Response:

For plants with a positive moderator temperature coefficient, those transients which generate a low low steam generator level signal, and result in an increase in core temperature, may have a [ ]+a,c increase in core power during the transient. The amount of time available to delay reactor trip is a function of initial power level: the lower the initial power level, the longer [

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]+a,c,e,f occurrence, the analyses presented in WCAP 11325 were performed with an initial power level associated with the interlock setpoint. For example, if the P-8 interlock setpoint was selected to be at 50% power, the supporting plant specific transient analysis would select the initial power level to be 50% of nominal plus uncertainties. This would bound a potential situation where an event was initiated at a lower power level, and due to a positive moderator temperature coefficient, resulted in an increase in power.

Question:

6. Various models of steam generators are in use by licensees, but only one model of steam generator was used for each loop configuration in the subject analysis. How will the characteristics of different steam generator models be accounted for in plant-specific implementation?

Response:

The supporting analyses, which were used to determine the allowable trip delays, were based upon a feedring type steam generator. Analyses to verify that the trip delays would not lead to a violation of Condition IV acceptance criteria were based upon both feedring and preheat type steam generators. In general, the transient heat transfer characteristics of Westinghouse steam generators, as they affect the primary side conditions, would be expected to be similar during a loss of heat sink event. This is evident from loss of feedwater ATWS analyses conducted as early as 1974, and reported in WCAP-8330. This WCAP was applicable for plants equipped with steam generator models 51 and D.

Since WCAP-11325 (P) and WCAP-11326 (NP) describe only the functional design and methodology of imposing power-dependent adjustable time delays to the low-low steam generator water level trip, the applicability would not be dependent upon the steam generator model. For further information concerning plant-specific implementation and applicability, see the response to Question 7.

Question:

7. WCAP-11325 (P) and WCAP-11326 (NP) provide analyses of the Loss of Normal Feedwater and the Feedline Break Accident to support trip delay. It also provides a general description of the logic to be used for the steam generator low level trip and auxiliary feedwater startup logic.

   (1) Describe in detail what information will be provided by licensees in support of use of these modifications at each plant in addition to, or in place of, that provided in WCAP-11325 (P) and WCAP-11326 (NP).

   (2) List the factors that will be used to determine whether the transient and accident analyses in WCAP-11325 (P) and WCAP-11326 (NP) will be applicable to (or bounding for) a particular reactor including possible differences in core, system and plant design.

Response:

(1) It is expected that the following items will be submitted to support an application to implement the adjustable trip delay modifications:

Plant-specific protection system logic diagrams will be submitted as revisions to the appropriate sections of the FSAR.

FSAR revisions will be provided, as necessary, to reflect the trip delay time addition and to assure that the limiting transients are represented among the accident analyses. Safety evaluations for the affected transients, as required by 10CFR50.59, and including an I & C safety evaluation will also be provided.

Changes to the plant-specific technical specifications will be recommended, similar to the example in WCAP-11325 (P) and WCAP-11326 (NP), and a Significant Hazards Evaluation, as required by 10CFR50.92, will also be provided.

Individual applications for approval to implement the WCAP-11325 (P) and WCAP-11326 (NP) delayed SG low-low level trip modifications would be expected to include information demonstrating that the imposition of the programmed trip delays would not invalidate the accident analysis conclusions contained in the plants' respective licensing bases. Plant-specific accident analyses or evaluations, protection system logic diagram revisions, and proposed technical specifications would be submitted, based upon the methods, results, and examples of WCAP-11325 (P) and WCAP-11326 (NP).

The affected accidents are those accidents during which the first reactor protection system signal generated is the low-low steam generator water level trip signal (e.g., loss of normal feedwater, loss of AC power and feedwater line rupture, if applicable). According to the protection system logic described in WCAP-11325 (P) and WCAP-11326 (NP), two permissives are used to interlock the steam generator low-low water level trip delays: P-7 and P-8. The P-7 permissive is set at approximately 10 percent power, and is used to permit a trip delay during transients affecting more than one steam generator. The P-8 permissive is set above 50 percent power in some plants, and is used to permit a trip delay during transients that affect only one steam generator. As the P-8 permissive setpoint is lowered, longer trip delays become justifiable. Therefore, it is possible to select a desired trip delay time by adjusting the interlocking P-8 permissive setpoint.

Once the trip delays are selected (approximately four minutes is recommended), and the P-8 permissive setpoint, which is used as a power interlock, will be determined and set. The associated power uncertainty in the P-8 setpoint will be verified and the P-8 interlock will be assumed in subsequent accident analyses or evaluations. It is expected that the P-8 setpoint will be lowered to make more delay time available. Lowering this permissive setpoint, should not have an adverse safety impact.

Plant-specific loss of feedwater accident analyses may be performed, considering the selected trip delays at their interlocking power levels. For some plants, it may be possible to apply WCAP-11325 (P) and WCAP-11326 (NP) to envelope the plant-specific accident analyses. In these cases, no accident analyses will be performed. Instead, evaluations will be made to show that the WCAP-11325 (P) and WCAP-11326 (NP) results are applicable to the particular plant design. Accident analyses would be required for two-loop plants, since WCAP-11325 (P) and WCAP-11326 (NP) contain results only for three and four-loop plants.

Loss of normal feedwater to only one steam generator, as well as the complete loss of normal feedwater will be considered in terms of SRP acceptance criteria. If necessary, FSAR revisions, consisting of applicable text, figures or references, will be provided in order to represent the most limiting case.

The feedwater line rupture accident will also be checked. Since this is a Condition IV event, the object is verify that there is adequate protection available after the the trip delays are incorporated. The feedwater line rupture accident will be analyzed or evaluated to show that the FSAR conclusions are not invalidated by the protection system modifications (if the feedwater line rupture accident is in the licensing basis). Results of WCAP-11325 (P) and WCAP-11326 (NP) will be used as much as possible.If necessary, FSAR revisions, consisting of applicable text, figures or references, will be provided in order to represent the most limiting case.

The plant-specific protection system logic diagrams will be revised to include the adjustable delays to the steam generator low-low water level trip signal. The revisions will be consistent with; but not necessarily identical to, the conceptual design provided in WCAP-11325 (P) and WCAP-11326 (NP). Appropriate FSAR revisions will be submitted.

Revisions to the plant-specific technical specifications will be recommended in order to account for the adjustable delays to the steam generator low-low water level trip signal.

Safety evaluations of the affected transients, if re-analysis is not required, and of the hardware modifications will be performed. A significant hazards evaluation (tech specs) will also be provided.

(2) The concept of the modifications described in the WCAP (i.e., trip delays that are set according to power level and the number of steam generators experiencing a low water level condition) would be [

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In order to judge whether the analyses of WCAP-11325 (P) and WCAP-11326 (NP) would be bounding for a particular plant, the plant's design parameters, operating conditions, and reactor protection system (logic and setpoints) would have to be compared to the corresponding assumptions used for WCAP-11325 (P) and WCAP-11326 (NP). The key points of comparison are summarized below:

Loop Configuration	3 or 4 loops
Power Rating	less than 2785 MWth or 3425 MWth
SG Model	F
Pressurizer Volume	1400 or 1800 cu ft
Pressurizer Relief	typical 3 and 4 loop capacities
Auxiliary FW Flow	typical 3 and 4 loop capacities
Auxiliary FW Enthalpy and Purge Volume	
Auxiliary FW System	typical 3 and 4 loop configurations

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These parameters, and the information provided in Tables 1 and 2 of WCAP-11325 (P) and WCAP-11326 (NP), are the basis for determining whether the transients of WCAP-11325 (P) and WCAP-11326 (NP) can be directly applied to a particular plant.

Question:

8. WCAP 11325 uses the ANSI/ANS-5.1 - 1979 decay heat model. The treatment of decay heat with the 1979 model is more complex than for the 1971 decay heat model required by 10 CFR 50, Appendix K. More options are left to the user with the 1979 model. For example, the user must supply Q values (Mev/fission) for the three fissile isotopes, a correction to the decay heat, G, to account for neutron captures in fission products and a power history for a finite operation time. Since the use of this model is intended to justify changes to the reactor protection system and plant safety analyses, all input assumptions should be specified. If any of these plant assumptions are to be plant specific, please specify this also.

Response:

A. FUEL -

The model uses the values for the decay heat power from uranium-dioxide pressurized light water reactor fuel. The model assumes a fuel load initially containing Uranium-235 and Uranium-238 and accounts for the generation of Plutonium-239 as a function of fuel burnup. The model bounds Standard, Optimized, and Vantage-5 Westinghouse fuel.

The contributions from Uranium-239 and Neptunium-239 are specifically accounted for as indicated in Section 4 of ANSI/ANS-5.1-1979.

A spectrum of fuel enrichments from 1.5 w/o U-235 to 5.0 w/o U-235 were evaluated.

B. POWER HISTORY -

The model assumes that, from the initial load conditions, the fuel is operated at a full power condition until the assumed shutdown time. Calculations were made for a range of region-average burnups of the fuel ranging from 1000 MWd/MTU to 50000 MWd/MTU with the specific range of region-average burnup dependent upon the weight percent of the fuel.

C. CAPTURE CORRECTIONS -

The model accounts for a burnup and enrichment dependence of the Uranium-238 capture to fission ratio used in calculating the Uranium-239 and Neptunium-239 decay heat power contribution.

The model accounts for the effect of Neutron Capture in Fission Products as described in Section 3.5 of ANSI/ANS-5.1-1979. Table 10 of ANSI/ANS-5.1-1979 is used for shutdown times of less than 10,000 seconds and Equation 11 of ANSI/ANS-5.1-1979 is used for shutdown times greater than 10,000 seconds.

D. SPATIAL DISTRIBUTION -

A spatial distribution for variation in decay heat power is not included in the model.

E. UNCERTAINTY ALLOWANCES -

The model utilizes Equations 5a, 5b, 9, and 10 of ANSI/ANS-5.1-1979 in the calculation of the decay heat uncertainty with the exception of the delta-P/P is assumed to be 0 in the model. This uncertainty (delta-P/P) is accounted for in the LOFTRAN code independently (if assumed) and therefore this power level uncertainty would not be convoluted into the total decay heat uncertainty.

The standard deviation in in decay heat power for Uranium-235, Uranium-238, and Plutonium-239 were taken from Tables 4, 6, and 5, respectively, in ANSI/ANS-5.1-1979.

The standard deviation calculated using the assumptions above was doubled to form the final uncertainty on decay heat power level.

F. Q VALUE (Mev/FISSION)

The assumed Mev/fission for all nuclides in the analysis was 200 Mev/fission.

G. DECAY HEAT CALCULATION -

a. For each combination of enrichment and burnup analyzed, the decay heat level were calculated for shutdown times from 1 to 80,000 seconds as a fraction of the fuel nominal power.
b. An integration of each set of data was used to generate a total decay heat contribution, in full power seconds, for each combination of enrichment and burnup.

c. For each shutdown time calculated, the maximum total decay heat contribution from any of the combinations of enrichment and burnup was selected for application in the LOFTRAN model.

d. A set of constants were selected which would model the decay heat levels in LOFTRAN to bound the maximum total decay heat contribution values selected above for up to 10,000 seconds (the time of interest for LOFTRAN analyses) following shutdown.

Question:

9. On page 32 the statement: "The potential reactor system cooldown resulting from a secondary pipe rupture is evaluated in a steamline break analysis" is made. Do any secondary pipe rupture analyses take credit for protective actions on low-low steam generator level? If so, what effect would the delay modifications to this trip have on these events?

Response:

A rupture of a secondary side pipe can result in an RCS heatup or an RCS cooldown. The purpose of a feedline rupture analysis is to evaluate the potential RCS heatup. The feedline rupture event was analyzed and reported in this WCAP. The potential RCS cooldown resulting from a secondary pipe rupture is evaluated in a steamline break analysis. Steamline break core response and mass and energy inside containment analyses take no credit for actuation of any protective function on a low-low steam generator water level signal.

Steamline breaks outside containment for small break sizes may use the low-low steam generator water level trip as the first credited trip signal. [

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Question:

10. On page 58 the first sentence of Section 4.6 discusses one or two timers. Should this section also include discussion for modifications up to four timers?

Response:

The purpose of Section 4.6 is to demonstrate that the operator will be made aware of the low-low level condition and that the system is in operation. A manual reactor trip is therefore not immediately required and that there is an opportunity to restore the steam generator water level and avoid the trip. Therefore, the discussions in Section 4.6 are applicable to any number of timers.

Question

11. On page 63 in the last sentence of the bottom paragraph, the cancelling of time delays and the immediate initiation of protection action are discussed. This statement appears incorrect in terms of a modification with four time delays. Please clarify.

Response:

In general, the WCAP was written with a basic two-timer system in mind. The two-timer system is expected to be the most-applied option. The statement on page 63 is incorrect in terms of a modification with four timers (as shown in Figure 25, page 61). For a four-timer system, [

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imposed. This, and other alternatives may be proposed to achieve the same effect, all of which would be consistent with the concept described in WCAP-11325 (P) and WCAP-11326 (NP), and would be supported by the same accident analyses.

Question:

12. On page 65 a general discussion of timer designs is provided. Please provide a detailed discussion of timer designs typical for each type of reactor protection system (e.g., relay, solid-state). The use of electronic timers rather than electro-mechanical types should be considered.

Response:

Final selection of time delay hardware has not yet been completed, however every attempt is being made to use a timer which has already been qualified for use in Class 1E nuclear control and protection systems. The preliminary design is based upon an AGASTAT 7000 series electropneumatic timing relay. This is a timing relay model which is already in use in Westinghouse-designed solid state protection systems (for the safety injection reset timer circuit). This timer model has been used for years in the solid state protection system with no history of problems or failures.

If the AGASTAT series 7000 timing relay cannot be used for this application, an alternate will be selected on the basis of:

* ability to perform correctly within the circuit design characteristics.

* previous seismic and environmental qualification credentials.

* ability to pass a seismic and environmental qualification test.

* previous performance history with respect to accuracy, repeatability, and reliability.

Use of electronic time delay devices will be considered during the selection process of the alternate.

The circuit design for the hardware upgrade shown in Figure 4 has been carefully developed to maximize system reliability and availability and to insure safety function capability and compliance to applicable regulatory guides.

As the schematic in Figure 4 demonstrates, each delay function T1 and T2 is incorporated using two parallel configured master relay/slave delay relay sets. The contact output of each set is then series wired into the 1/2 voting logic circuit for reactor trip and auxiliary feedwater actuation. This design philosophy was used to overcome any potential loss of system reliability or safety function due to hardware or power source failure.

* Hardware Failure; master or slave delay relays - each set of master/slave delay relays is maintained in a normally energized mode during system operation. If a relay failure causes either a master or slave relay from any set to de-energize, the associated slave relay timer will time itself out and the associated contacts will close. However, the system will continue to function on the remaining set of master/slave delay relays with no inadvertent reactor trip or loss of safety function.

* Power Source Failure - the use of 48VDC to energize both the master and slave delay relays in each set provides auctioneered 48VDC safety power from two channel supplies for each train. Any loss of a single channel power source will have no adverse effect on the reliability of safety capability of that solid state protection system train.

* Periodic Testing - regularly scheduled on line testing of the complete solid state protection system operation insures that there are no undetectable relay failures.

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Figure 4:
Hardware Schematic for Delayed Trip

Question:

13. On page 65 a general discussion of timer setpoint adjustment is provided. Please provide a discussion od proposed technical specifications for these adjustments based upon their similarity to steam generator low-low level trip setpoint (i.e., they also impact validity of safety analyses).

Response:

The adjustment of the time delays discussed on page 65 will be accommodated within the proposed technical specifications indicated on page 2 of Appendix II. The total response times shown on page 2 would include any errors due to the accuracy of the timers. The verification of the response times in these proposed technical specifications can be handled in the same way as the 2 second response time is currently handled for the low low level trip. That is, the response time required in Table 3.3-2 of the technical specifications would comply with technical specification 4.3.1.1 and 4.3.1.2 to demonstrate both the operability and total response time of the trip function.

Question:

14. On page 66 a general discussion of system testing is provided. Using schematics or block diagrmas for typical reactor protection systems, provide a detailed discussion of testing for the modified systems with timers."

Response:

Final test designs, procedures and instructions have not yet been completed, however any and all final testing will utilize the same test philosophy as that which is already established for on-line periodic testing of solid state protection system plants.

A preliminary test design of this upgrade is as follows:

On line testing of the hardware upgrade for the modified steam generator low-low water level delayed reactor trip will be done in two stages.

Stage one will test the operation of the logic circuit additions between test points TP1 and TP2 shown in Figure 4. This test will use the semi-automatic tester circuit to pulse through the added logic and determine if the circuits are functioning properly. The test will not cycle the master/slave timer relay sets. Additional test positions will be added to the semi-autmatic tester switches to test this logic.

Stage two will test the operation of the master/slave timer relay sets (beginning at test point TP-2) and the remaining reactor trip and safeguards actuation logic. Overlapping testing will be maintained between test states one and two.

As part of this reactor protection logic upgrade, the safeguards test cabinet test circuits will be maintained as go tests to comply with existing tech spec and FSAR requirements regarding auxiliary feedwater testing.

Testing during this stage will require that the reactor trip bypass breakers be racked in and closed. Separate test switches located on the master relay test panel will de-energize the master relays which in turn will cause the slave timer relays to time out and trip thereby causing a simulated reactor trip and auxiliary feedwater safeguards actuation.

A complete test procedure will be provided as part of the manual revision for the solid state protection system for this modification. In addition, a revised test procedure for the safeguards test cabinet will also be provided.

Question:

15. On page I-7 the logic configurations for the P-7 and P-8 interlocks are discussed. In light of recent problems with the P-10 interlock (see IE Information Notice No. 86-105), discuss the consequences of interlocking additional reactor trips and protection actions with interlocks having 3/4 logic. Also discuss any technical specification change that may be desirable to reduce the vulnerability of interlocks with 3/4 logic to single failures.

Response:

If there is a failure in the P-7 or P-8 permissive interlocks similar to the P-10 permissive failure discussed in IE Information Notice No. 86-105, the timers will not be actuated, and no delay will be imposed upon the reactor trip and auxiliary feedwater startup demanded by the low-low steam generator water level trip signal. (If four timers are used, then a short delay may still result; but this delay is not dependent upon the permissive interlocks, and is fully supported by accident analyses.)

The proposed modification to the Steam Generator level logic for incorporating a timer circuit interlocked with the P-7 and P-8 permissive interlocks is not considered to be susceptible to the concerns which were raised with the P-10 interlock for the following reasons:

1. Removal of the P-10 interlock <u>enables</u> reactor trip circuits for:

A. Power Range High Neutron Flux (low setpoint)

B. Intermediate Range High Neutron Flux

C. Source Range High Neutron Flux (when P-6 has reset)

Additionally, removal of the P-10 interlock enables the Source Range channels by connecting high voltage to their detectors.

Accident analyses take credit for the above reactor trips. Thus, the Protection System must satisfy IEEE Std. 279-1971 single failure criterion. However, in order to remove the P-10 interlock, it is necessary for 3/4 Power Range flux channels to be below their setpoint. In the case where one channel is out service (tripped), a single failure to a second channel will prevent removal of the P-10 interlock since the required coincidence cannot be satisfied with only the two remaining channels. In this case, the required reactor trips will not be enabled which was the cause for the P-10 concern.

2. Unlike the P-10 interlock, the P-7 and P-8 interlocks are used to disable reactor trip circuits. Reactor trips which are disabled by removal of the P-7 interlock are:

   A. Reactor Coolant Low Flow

   B. Pressurizer Low Pressure

   C. Pressurizer High Water Level

   The Reactor trip which is disabled by removal of the P-8 interlock is Reactor Coolant Low Flow

In the case where one channel is out of service (tripped), a single failure to a second channel will prevent removal of the affected interlock (P-7 or P-8). However, the only consequence is that the affected reactor trips will not be disabled. This is considered to be conservative from a safety standpoint since required reactor trips will not be prevented.

The only circuit enabled by removal of the P-7 and P-8 interlocks is the proposed timer circuit for delay of the Steam Generator low-low level reactor trip. The functional change in this area required by the proposed modification to the Steam Generator level logic is that failure to remove either the P-7 or P-8 interlock will only prevent enabling of the timer circuit and additional time delay for the reactor trip. This is considered to be conservative from a safety standpoint since a reactor trip will be actuated sooner than it is actually required

Accordingly, technical specification changes to reduce the vulnerability of the P-7 and P-8 interlocks to single failures are not planned.

Also, In WCAP 11325, page I-7, it is stated that the input signals required to remove interlock P-7 are 3/4 Neutron Flux (Power Range) below setpoint (from P-10) or 2/2 turbine impulse chamber pressure below setpoint (from P-13). It should be stated that the input signals required to remove interlock P-7 are 3/4 Neutron Flux (Power Range) below setpoint (from P-10) and 2/2 turbine impulse chamber pressure below setpoint (from P-13).

Question:

16. On Page I-10 in Section 7.7.6, the closure of blowdown and sample isolation valves is discussed. Please discuss what plants have this closure on low-low steam generator level and what effect of delaying the closure of these valves has on accidents or transient analyses.

Response:

The purpose of these valves is to remove sludge buildup from the steam generators. The valves are designed to operate at a typical liquid mass flow rate of [ ]+a,c. They do have the capacity to operate at higher flow rates under limited conditions if samples taken indicate a higher than normal sludge content or if there is an effort to clean out the steam generators (i.e., just prior to a refueling shutdown). Conservative estimates indicate that the blowdown isolation and sample isolation valves would operate [ ]+a,c.

All Westinghouse plants have the closure of the blowdown isolation and sample isolation valves function on low-low steam generator level. These valves are safety-grade equipment, and their closure is required when auxiliary feedwater starts, since the auxiliary feedwater system is sized based on the assumption that these valves are closed.

At full power, the maximum time delay found in the analyses to be tolerable is less than [ ]+a,c There are sufficient conservatisms in the steam generator masses assumed in the analyses to offset any inventory lost during the additional time delays. At reduced power levels, the auxiliary feedwater system has capacity in excess of what is needed to remove decay heat at the lower power levels, since the auxiliary feedwater system was sized based on results of Engineered Safeguards Design power calculations.

Question:

17. Instrumentation uncertainties used in setpoint determinations take into account harsh environmental conditions. Assumptions about the timing of radiation, temperature and pressure during an accident or transient may influence the values of uncertainties used. In view of this, discuss the impact, if any, upon instrumentation uncertainties associated with a harsh environment caused by delaying the actuation of reactor trip and auxiliary feedwater.

Response:

Delaying the reactor trip and actuation of auxiliary feedwater, demanded by low-low steam generator water level conditions, could affect the harsh environment caused by steamline or feedwater line rupture accidents, only if these protective functions are assumed to be initiated by the low-low steam generator water level trip signal (i.e., reactor trip and auxiliary feedwater are not initiated by another signal, such as low steamline pressure SI, before the low-low steam generator water level trip signal is generated). If such steamline or feedwater line rupture accidents can be postulated, then it must be determined whether the potential effect upon the resultant harsh environment would aggravate (i.e., exceed) the currently assumed environment that is used for the calculation of instrument uncertainties (if applicable, depending upon the plant licensing basis).

Inside containment, the only postulated accident that relies upon the steam generator low-low water level trip signal to trip the reactor and actuate auxiliary feedwater, and thereby possibly affect the containment environment through the preset delay, would be the feedwater line rupture. Other accidents that could produce an adverse environment, such as the steamline rupture, do not rely upon the low-low steam generator water level trip signal as the primary initiator of protective functions. The worst-case environmental conditions that are used to qualify instruments inside containment are based upon a combination of conditions caused by postulated LOCA, steamline rupture, and feedwater line rupture accidents. The highest temperatures are produced by the postulated steamline rupture cases, and analyses of these cases indicate that the reactor trip would always be generated by conditions other than low steam generator water level (e.g., low steamline pressure or high containment pressure). Since the mass release from a steamline rupture would be at a higher enthalpy than any corresponding mass release from a feedwater line rupture, and since these mass releases are of approximately equal duration, it is judged that the energy released from a feedwater line rupture (delay included), inside containment, would be less than the energy released from a steamline rupture.

Outside containment, the environment produced by a feedwater line rupture (delay included) is also judged to be less severe than the environment produced by a steamline rupture, for the same reasons that the feedwater line rupture inside containment was judged to be less limiting than the steamline rupture, due mainly to the lower enthalpy of the steam generator mass release. [

]+a,c

]+a,c

Therefore, it is judged that delaying the steam generator low-low water level trip would not significantly affect instrumentation uncertainty due to harsh environmental conditions.