JUN 28 1985

TO ALL APPLICANTS AND LICENSEES WITH WESTINGHOUSE (W) DESIGNED NUCLEAR STEAM SUPPLY SYSTEMS (NSSSs)

SUBJECT: IMPLEMENTATION OF TMI ACTION ITEM II.K.3.5, "AUTOMATIC TRIP OF REACTOR COOLANT PUMPS" (GENERIC LETTER NO. 85-12)

Gentlemen:

The purpose of this letter is to inform you of (1) the staff's conclusions regarding the Westinghouse Owners Group (WOG) submittals on reactor coolant pump trip in response to Generic Letters 83-10c and d, and (2) provide guidance concerning implementation of the reactor coolant pump trip criteria. Our Safety Evaluation (SE) on this subject is enclosed for your use.

With regard to the WOG submittals referenced in Section V of the enclosed SE, we conclude that the methods employed by the WOG to justify manual reactor coolant pump (RCP) trip are consistent with the guidelines and criteria provided in Generic Letters 83-10c and d. The approved Westinghouse Small Break LOCA Evaluation Model was used to demonstrate compliance with 10 CFR 50.46 and Appendix K to 10 CFR Part 50.

We have determined that the information provided by the WOG in support of the alternative RCP trip criteria is acceptable on a generic basis. A suitable reactor coolant pump trip criterion can be selected by each licensee to minimize reactor coolant pump trip during steam generator tube ruptures and non-LOCA events, while still providing for RCP trip for small break LOCAs.

With regard to implementation, we note that the WOG RCP trip methodology allows applicants/licensees to select among three alternate RCP trip criteria. The selection is based upon obtaining maximum discrimination between a small break LOCA (which requires RCP trip) and a steam generator tube rupture (which does not require RCP trip). In reviewing the WOG RCP trip criteria, we note that the process of criterion selection involves a number of considerations which were assigned plant-specific status by the WOG during the process of the trip criteria review.

Accordingly, we request that operating reactor licensees select and implement an appropriate RCP trip criterion based upon the WOG methodology. Schedules for submittal of information requested in Section IV of the SE (refer to Appendix A for considerations associated with Generic Letters 83-10c and d) should be developed with your individual project managers within 45 days from receipt of this letter. The requested information does not constitute a new

CONTACT:
D. Jaffe
X28140

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requirement but only identifies information specified in Generic Letters 83-10c and d which has not been provided under the WOG generic program. In the event that licensees decide not to trip the RCP (an option provided for in Generic Letters 83-10c and d), they should respond to the questions in Section IV of the SE and refer to Appendix B of the SE. Applicants should provide the appropriate response to the extent that this information is known at this time.

Those applicants and licensees who choose not to endorse the WOG methodology should submit a schedule for submittal of plant specific RCP trip criteria or justification for non-trip of RCPs within 45 days of receipt of this letter.

This request for information was approved by the Office of Management and Budget under clearance number 3150-0065 which expires September 30, 1985. Comments on burden and duplication may be directed to the Office of Management and Budget, Reports Management, Room 3208, New Executive Office Building, Washington, D.C. 20503.

If you believe further clarification regarding this issue is necessary or desirable, please contact Mr. D. Jaffe (301 492-8140).

Sincerely,

Original Signed by
Hugh L. Thompson, Jr.

Hugh L. Thompson, Jr., Director
Division of Licensing

Enclosure:
Safety Evaluation

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Service Lists

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Those applicants and licensees who choose not to endorse the WOG methodology should submit a schedule for submittal of plant specific RCP trip criteria or justification for non-trip of RCPs within 30 days.

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If you believe further clarification regarding this issue is necessary or desirable, please contact Dr. B. Sheron (301 492-7460).

Sincerely,

Hugh L. Thompson, Jr., Director
Division of Licensing

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ORB#3/DL
G.Holahan
6/27/85

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION FOR THE WESTINGHOUSE OWNERS GROUP REACTOR COOLANT PUMP TRIP

I. INTRODUCTION

TMI Action Plan Item II.K.3.5 of NUREG-0737 required all licensees to consider other solutions to the small-break loss-of-coolant-accident (LOCA) problems because tripping the reactor coolant pumps (RCPs) was not considered the ideal solution. Automatic trip of the RCPs in the case of a small-break LOCA was recommended until a better solution was found. A summary of both the industry programs and the NRC programs concerning RCP trip is provided in Generic Letters 83-10a, b, c, d, and e, which are included in the NRC report, SECY-82-475, from W. J. Dircks to the NRC Commissioners, "Staff Resolution of the Reactor Coolant Trip Issue" (November 30, 1982). SECY-82-475 also provided the NRC guidelines and criteria for the resolution of TMI Action Item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps."

In SECY-82-475 the NRC concluded: "...that appropriate pump trip setpoints can be developed by the industry that would not require RCP trip for those transients and accidents where forced convection circulation and pressurizer pressure control is a major aid to the operators, yet would alert the operators to trip the RCPs for those small LOCAs where continued operation or delayed trip might result in core damage."

SECY-82-475 also stated: "The resolution provided in the enclosures [Generic Letters 83-10] is intended to ensure that for whatever mode of pump operation a licensee elects, a) a sound technical basis for that decision exists, b) the plant continues to meet the Commission's rules and regulations, and c) as a minimum, the pumps will remain running for those non-LOCA transients and accidents where forced convection cooling and pressurizer pressure control would enhance plant control. This would include steam generator tube ruptures (SGTR) up to approximately the design basis event (one tube)."

The Westinghouse Owners Group (WOG) submitted two reports to the NRC in response to the Westinghouse specific Generic Letters, 83-10c and d. The first report provided an "Evaluation of Alternate RCP Trip Criteria" (Reference 1). The second report provided the "Justification of Manual RCP Trip for Small Break LOCA Events" (Reference 2). The WOG also provided additional information (Reference 3) in response to our request for this information, based on the review of the WOG submittals. We have also performed analyses of selected events to support our review (Reference 4).

Appendix A to this report summarizes Section I of the enclosure to Generic Letter 83-10 for "Pump-Operation Criteria that Can Result in RCP Trip During Transients and Accident," and Appendix B summarizes Section II, "Pump-Operation Criteria That Will Not Result in RCP Trip During Transients and Accident."

II. SUMMARY

The WOG has developed a set of three alternative reactor coolant pump (RCP) trip criteria - each one being reported to be equally suitable in meeting the intent of Generic Letter 83-10. The revised criteria replace the current RCP trip criterion of low reactor coolant system (RCS) pressure, which could result in RCP trip for SGTR and non-LOCA events.

The objective of the WOG study was to evaluate alternative RCP trip criteria to determine if a criterion could be established to reduce the probability of RCP trip for SGTRs and non-LOCA events, while still providing for RCP trip for small break LOCAs.

The parameters which were considered for the evaluation of alternative RCP trip criteria included RCS pressure, RCS subcooling, and secondary pressure dependent RCS pressure (RCS/secondary pressure differential). Because SGTRs and most non-LOCA events will not result in adverse containment conditions, the normal instrument uncertainties associated with the measurement of these parameters can be utilized in evaluating the effectiveness of these alternative criteria in preventing pump trip for most SGTRs and non-LOCAs.

The alternative RCP trip criteria which were evaluated are:

1. RCS pressure with normal instrument uncertainties.

   This criterion would be established in the same manner as the current RCS pressure criterion, with the exception that the normal instrument uncertainties would be utilized in determining the RCP trip setpoint for normal containment conditions. The instrument uncertainties associated with post-accident containment conditions would continue to be used to determine the RCP trip setpoint for adverse containment conditions.

2. Reactor coolant subcooling.

   This method would provide a direct indication of the need for pump trip, since pump trip is not required as long as the reactor coolant remains subcooled. The RCP trip setpoint would be established as zero degrees subcooling in the RCS hot legs, plus the uncertainty in the subcooling monitor to assure that the pumps are tripped before subcooling is actually lost. The normal instrument uncertainties would be used for normal containment conditions, whereas the instrument uncertainties associated with post-accident containment conditions would be used for adverse containment conditions.

3. Secondary pressure dependent RCS pressure.

With the current method of using RCS pressure, the trip criteria is conservatively derived assuming that the secondary pressure is at the lowest secondary safety valve set pressure. However, the secondary pressure may actually be significantly less than this value, particularly if the condenser steam dump system is in operation. With this method, the RCS pressure setpoint for pump trip would be continuously evaluated based on the actual secondary pressure. The alternate RCP trip criterion can also be expressed as the RCS/secondary pressure differential. The combined instrument uncertainties for the RCS and secondary pressure measurements would be included in determining the RCP trip setpoint. The normal instrument uncertainties would be used for normal containment conditions, whereas the instrument uncertainties associated with post-accident containment conditions would be used for adverse containment conditions.

The results of the small break LOCA analysis demonstrate that each of the alternative criteria is adequate in providing an indication for the operator to trip the RCPs for small break LOCAs. Thus, the selection of the RCP trip criterion can be based on the capability to preclude a pump trip for SGTRs and non-LOCAs. The minimum values of each of the parameters used to evaluate the alternative criteria were also determined for SGTRs and non-LOCA events for each category of plants in the study. A methodology was provided to determine the RCP trip setpoints for each of the three alternative criteria. Using the results of the evaluation and the setpoints calculated for each of the alternative criteria, each utility can determine which of the criteria will prevent pump trip for SGTRs and non-LOCA events for their respective plants. The criterion that is considered most appropriate in providing pump trip discrimination between LOCAs and SGTR or non-LOCA events can then be selected by the utility for each plant.

Based on these studies, the WOG concluded that the RCP trip criterion can generally be implemented using existing qualified instrumentation already available in the plants, and additional instrumentation is not required.

The WOG followed the guidelines provided in Generic Letter 83-10c and d to justify manual RCP trip for small-break LOCAs. (See Appendix A, Section D.) The WOG studies concluded that:

1. Every Westinghouse plant's FSAR ECCS analysis demonstrates compliance with 10 CFR 50.46 if operator action to trip the RCPs is taken within two minutes after the RCP trip criterion is reached.

2. Most probable best estimate analyses indicate that in all Westinghouse plants the RCPs may be tripped at any time during a small break LOCA event without reaching clad temperatures of 2200°F. The highest PCT calculated with most probable best estimate assumptions was 1255°F.

The WOG concluded that automatic reactor coolant pump trip is not required since adequate time for manually tripping the RCPs is demonstrated using 10 CFR Part 50, Appendix K assumptions as well as most probable best estimate analysis results. It was also concluded that the most probable best estimate analysis results demonstrate that the RCPs can be tripped at any time during the LOCA (if the operator should fail to trip the pumps when the trip criterion is reached) without incurring unacceptable clad temperature results. Therefore, the WOG concludes that the existing guidelines in Revision 1 of the Emergency Response Guidelines (ERGs) are sufficient and complete with respect to RCP status for all accident situations, and no additional "missed RCP trip setpoint" steps are required.

The methods (References 5 and 6) employed by the WOG to justify manual RCP trip are consistent with the guidelines and criteria provided in Generic Letters 83-10c and d.

We have reviewed the assumptions and models employed by the WOG to study steam generator tube rupture (SGTR) and non-LOCA events. The LOFTRAN computer program (Reference 7) was used to study these events. Best estimate assumptions and models were used. However, the SGTR break flow model incorporated into LOFTRAN does result in higher than expected mass flow rates for a given break size. (This model was approved for SAR SGTR analyses, where the high flow rates result in a conservative evaluation of offsite dose.) The WOG position with respect to the use of this conservative model is that the analysis results are bounding for the design basis SGTR event of a single tube.

The WOG considered all other FSAR Chapter 15 non-LOCA events for evaluation against the alternate RCP trip criteria. It was concluded that the feedline and steamline breaks needed to be considered because their transient characteristics would be the most limiting with respect to the three criteria. For the steamline break accident, a "credible" (4.5 inch diameter) break size was considered - essentially equivalent in size to a single steam generator PORV failing open. For the feedline break, a full double-ended rupture of the main feedwater line was considered.

We have reviewed the assumptions, models and plant groupings used to perform the SGTR and non-LOCA studies and have determined that the information provided is acceptable. Table 1 provides a summary of the WOG studies. We believe that the three RCP trip criteria may be marginal for some plants for the SGTR event. We base our conclusion on the following:

1. The SGTR event gives the minimal values for all three alternative RCP trip criteria for all but a few plants, and

2. The uncertainty analysis of instrument error provided by the WOG for use to evaluate the trip set points for each alternative criteria (for both normal and adverse containment conditions) may not be bounding for all plants.

In particular, the reactor coolant system pressure set point RCP trip criterion appears to offer the least in reducing unnecessary RCP trip. This confirms our position as discussed in SECY-82-475 and Generic Letter 83-10.

The WOG objective for the SGTR and non-LOCA analyses was to consider design basis accidents with more realistic assumptions, to enable the development of a RCP trip criterion which would provide reasonable assurance of continued pump operation for these accidents. While it is possible that other accident conditions could result in more limiting parameter values, the design basis accidents which were defined for the analyses combined with the conservatisms which are incorporated in the analytical model provide assurance that the analysis results will be bounding for most SGTR and non-LOCA events. The WOG does not consider it to be practical or necessary to develop a RCP trip criterion which will provide for continued pump operation for all possible SGTR and non-LOCA events. It would not be a safety problem if RCP trip should occur for a SGTR or non-LOCA event, since the plant safety systems are designed to handle those accidents with a loss of offsite power and, therefore, with RCP trip. The objective was to demonstrate that the RCPs will remain on for most of the expected cases of these accidents, so that the operator can retain normal pressurizer pressure control and will not be required to open the pressurizer PORVs. In addition, maintaining forced reactor coolant system flow will reduce the likelihood of generating voids in the reactor vessel upper head region.

The WOG response to our concern that none of the three alternative criteria would prevent RCP trip for a SGTR or a non-LOCA event (on a plant specific bases) is the recommendation to use the criterion which demonstrates the greatest discrimination capability.

In doing so, the WOG expects that a large range of SGTRs and non-LOCA events still would not require RCP trip. In the event of RCP trip occurring for SGTRs and non-LOCAs, the WOG position is that the Emergency Response Guidelines (ERGs) provide specific contingency actions to recover the plant even though RCP operation is not available. Also, specific RCP restart steps are built into the ERGs where deemed beneficial although they are not required for safe plant shutdown. The WOG expects, however, that at least one of the alternative criteria will be successful in preventing pump trip for SGTRs and non-LOCA events for each of the plants.

The studies performed by the WOG to determine the transient characteristics for the SGTR and non-LOCA events were based on best estimate input assumptions and models (to the extent practical with the computer programs used). Based on our experiences, with other thermal-hydraulic programs used to perform similar types of analyses, we believe there are uncertainties associated with the numerical results of any calculated system transient. Each licensee must consider these uncertainties when selecting the criterion which demonstrates the greatest discrimination capability, and be prepared to explain how they were considered during future inspections.

The generic nature of the WOG submittals concerning RCP trip by nature do not include any plant specific information, other than that needed to determine plant groupings for analysis. We have therefore included a section (Implementation) in this report which describes those plant specific items we require each licensee to address when incorporating the RCP trip criteria into the plant procedures.

III CONCLUSIONS

We have determined that the information provided by the WOG for the justification of manual reactor coolant pump trip is acceptable. The methods employed by the WOG to justify manual reactor coolant pump trip are consistent with the guidelines and criteria provided in Generic Letters 83-10c and d. The approved Westinghouse Small Break LOCA Evaluation Model was used to demonstrate compliance with 10 CFR 50.46 and Appendix K to 10 CFR Part 50.

We have determined that the information provided by the WOG in support of the alternative reactor coolant pump trip criteria is acceptable. A suitable reactor coolant pump trip criterion can be selected by each licensee to minimize reactor coolant pump trip during steam generator tube ruptures and non-LOCA events, while still providing for RCP trip for small break LOCAs.

The results presented by the WOG, for the plant groups studied, imply that one of the alternative RCP criteria would prevent RCP trip for the design basis SGTR and for design basis non-LOCA events. This would be a true statement if the numerical results from the calculation performed were error free and if each plant responds exactly as the simulation model predicts. Also, the uncertainty analysis for instrument error would have to be bounding for each plant, with normal containment conditions. Adverse containment conditions are not expected for design basis SGTRs or non-LOCA events.

We believe the analysis tools employed by the WOG are capable of qualitatively providing the appropriate information to evaluate the alternate RCP criteria. It should be obvious however that the quantitative values provided cannot be considered absolute. In our judgement, the alternate RCP trip criteria, as defined, may provide only marginal assurance of preventing RCP trip for the design base SGTR event for some Westinghouse plants.

We have concluded that the WOG has developed acceptable criteria for tripping the reactor coolant pumps during small-break LOCAs and to minimize reactor coolant pump trip for SGTR and non-LOCA events.

IV IMPLEMENTATION

The generic information presented by the WOG does not address plant specific concerns about instrumentation uncertainties, potential reactor coolant pump problems and operator training and procedures as requested in Generic Letter 83-10. Appendix A contains a summary related to these issues and may be used as a guideline to assure that these issues are adequately addressed.

In order to complete the response to Generic Letters 83-10c and d, each W licensee is required to submit the following information to the NRC for plant specific reviews:

A. Determination of RCP Trip Criteria

1. Identify the instrumentation to be used to determine the RCP trip set point, including the degree of redundancy of each parameter signal needed for the criterion chosen.

2. Identify the instrumentation uncertainties for both normal and adverse containment conditions. Describe the basis for the selection of the adverse containment parameters. Address, as appropriate, local conditions such as fluid jets or pipe whip which might influence the instrumentation reliability.

3. In addressing the selection of the criterion, consideration to uncertainties associated with the WOG supplied analyses values must be provided. These uncertainties include both uncertainties in the computer program results and uncertainties resulting from plant specific features not representative of the generic data group.

   If a licensee determines that the WOG alternative criteria are marginal for preventing unneeded RCP trip, it is recommended that a more discriminating plant-specific procedure be developed. For example, use of the NRC-required inadequate-core-cooling instrumentation may be useful to indicate the need for RCP trip. Licensees should take credit for all equipment (instrumentation) available to the operators for which the licensee has sufficient confidence that it will be operable during the expected conditions.

B. Potential Reactor Coolant Pump Problems

1. Assure that containment isolation, including inadvertent isolation, will not cause problems if it occurs for non-LOCA transients and accidents.

   a. Demonstrate that, if water services needed for RCP operations are terminated, they can be restored fast enough once a non-LOCA situation is confirmed to prevent seal damage or failure.

   b. Confirm that containment isolation with continued pump operation will not lead to seal or pump damage or failure.

2. Identify the components required to trip the RCPs, including relays, power supplies and breakers. Assure that RCP trip, when determined to be necessary, will occur. If necessary, as a result of the location of any critical component, include the effects of adverse containment conditions on RCP trip reliability. Describe the basis for the adverse containment parameters selected.

C. Operator Training and Procedures (RCP Trip)

1. Describe the operator training program for RCP trip. Include the general philosophy regarding the need to trip pumps versus the desire to keep pumps running.

2. Identify those procedures which include RCP trip related operations:

   (a) RCP trip using WOG alternate criteria
   (b) RCP restart
   (c) Decay heat removal by natural circulation
   (d) Primary system void removal
   (e) Use of steam generators with and without RCPs operating
   (f) RCP trip for other reasons

V REFERENCES

1. Westinghouse Owners Group, Letter OG-110, "Evaluation of Alternate RCP Trip Criteria," December 1, 1983.

2. Westinghouse Owners Group, Letter OG-117, "Justification of Manual RCP Trip for SBLOCA Events," March 9, 1984.

3. Westinghouse Owners Group, Letter OG-137, "Response to NRC Question on RCP Trip," October 25, 1984.

4. Lime, J. F., "TRAC Analysis of Small-Break and Tube-Rupture Accidents for the Evaluation of Westinghouse Alternate Reactor-Coolant-Pump Trip Criteria," Los Alamos National Laboratory, February 1985 (DRAFT).

5. Esposito, V., et al., "WFLASH - A FORTRAN IV Computer Program for Simulation of Transients in a Multi-Loop PWR," WCAP-8200 Rev. 2, June 1984.

6. Bordelon, F. M., et al., "LOCTA - IV Program: Loss-of-Coolant Transient Anaylsis," WCAP-8301, June 1984.

7. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907-P-A, April 1984.

Attachments:
Appendix A
Appendix B

TABLE 1 SUMMARY TABLE OF WOG REACTOR COOLANT PUMP TRIP STUDIES

PLANT	NO. LOOP	DOCKET NO.	OR TAC NO.	POWER LEVEL (MWT)	SI CUT OFF HEAD	PUMP WEIR Y/N	UHI Y/N	ACC. GAS PRES (PSIG)	UP HEAD TEMP H/C	S.G. VALVE SET PT (PSIA)	S.G. TUBE I.D. (IN)	NO LOAD TAVE (F)	MINIMUM RCS PRESSURE ---- (PSIA) ---- SGTR	SLB	FLB	MINIMUM RCS SUBCOOLING ----- (F) ---- SGTR	SLB	FLB	MIN RCS/SECONDARY DIFF. PRESSURE ---- (PSI) ----- SGTR	SLB	FLB
VOGTLE 1	4	50-424	0	3411	HI	Y	N	600	C	1200	.600	557	1753	1774	1868	58	71	69	685	811	817
VOGTLE 2	4	50-425	0	3411	HI	Y	N	600	C	1200											
SEABROOK 1	4	50-443	0	3411	HI	Y	N	600	C	1200											
SEABROOK 2	4	50-444	0	3411	HI	Y	N	600	C	1200											
MILLSTONE 3	4	50-423	0	3411	HI	Y	N	600	C	1200											
WOLF CREEK	4	50-482	0	3411	HI	Y	N	600	C	1200											
CALLAWAY	4	50-483	0	3411	HI	Y	N	600	C	1200											
BYRON 1	4	50-454	0	3411	HI	Y	N	600	C	1200	.664	557	1698	1774	1868	57	71	69	669	811	817
BYRON 2	4	50-455	0	3411	HI	Y	N	600	C	1200											
BRAIDWOOD 1	4	50-456	0	3411	HI	Y	N	600	C	1200											
BRAIDWOOD 2	4	50-457	0	3411	HI	Y	N	600	C	1200											
MC GUIRE 1	4	50-369	49653	3411	HI	Y	Y	400	C	1200											
MC GUIRE 2	4	50-370	0	3411	HI	Y	Y	400	C	1200											
CATAWBA 1	4	50-413	0	3411	HI	Y	Y	600	C	1200											
CATAWBA 2	4	50-414	0	3411	HI	Y	Y	600	C	1200											
MARBLE HILL 1	4	50-546	0	3411	HI	Y	N	600	C	1200											
MARBLE HILL 2	4	50-547	0	3411	HI	Y	N	600	C	1200											
WATTS BAR 1	4	50-390	0	3411	HI	Y	Y	400	C	1200											
WATTS BAR 2	4	50-391	0	3411	HI	Y	Y	400	C	1200											
COMANCHE PEAK 1	4	50-445	0	3411	HI	Y	Y	400	C	1200											
COMANCHE PEAK 2	4	50-446	0	3411	HI	Y	N	600	C	1200											
TROJAN	4	50-344	49697	3391	HI	Y	N	600	H	1100	.775	557	1526	1774	1868	52	71	69	535	811	817
ZION 1	4	50-295	49689	3250	HI	N	N	600	H	1100	.775	547	1497	1774	1868	61	71	69	604	811	817
ZION 2	4	50-304	49684	3250	HI	N	N	600	H	1100											
DIABLO CANYON 2	4	50-323	0	3411	HI	Y	N	600	H	1100	.775	547	1473	1774	1868	57	71	69	564	811	817
SALEM 1	4	50-272	49669	3411	HI	Y	N	600	H	1100											
SALEM 2	4	50-311	49670	3411	HI	Y	N	600	H	1100											
SEQUOYAH 1	4	50-327	49694	3411	HI	N	Y	400	C	1100											
SEQUOYAH 2	4	50-328	51528	3411	HI	N	Y	400	C	1100											

TABLE 1 (CON'T) SUMMARY TABLE OF WOG REACTOR COOLANT PUMP TRIP STUDIES

PLANT	NO. LOOP	DOCKET NO.	OR TAC NO.	POWER LEVEL (MWT)	SI CUT OFF HEAD	PUMP WEIR Y/N	UHI Y/N	ACC. GAS PRES (PSIG)	UP HEAD TEMP H/C	S.G. VALVE SET PT (PSIA)	S.G. TUBE I.D. (IN)	NO LOAD TAVE (F)	MINIMUM RCS PRESSURE ---- (PSIA) ---- SGTR	SLB	FLB	MINIMUM RCS SUBCOOLING ----- (F) ---- SGTR	SLB	FLB	MIN RCS/SECONDARY DIFF. PRESSURE ---- (PSI) ----- SGTR	SLB	FLB
D C. COOK 1	4	50-315	49701	3250	HI	N	N	600	H	1100	.775	547	1443	1774	1848	55	71	69	542	811	817
D.C. COOK 2	4	50-316	49702	3391	HI	N	N	600	H	1100											
DIABLO CANYON 1	4	50-275	49654	3338	HI	Y	N	600	H	1100											
SOUTH TEXAS 1	4	50-498	0	3800	IN	Y	N	600	H	1300	.664	567	1607	1830	1559	40	66	62	453	614	557
SOUTH TEXAS 2	4	50-499	0	3800	IN	Y	N	600	H	1300											
INDIAN POINT 2	4	50-247	49672	2758	LO	Y	N	600	H	1100	.775	547	1190	1955	1857	31	90	90	293	1050	986
INDIAN POINT 3	4	50-286	49687	3025	LO	Y	N	600	H	1100	.775	547	1211	1955	1857	32	90	90	315	1050	986
SUMMER	3	50-395	49115	2775	HI	Y	N	600	C	1200	.664	557	1543	1436	1923	51	70	71	549	790	822
SHERON HARRIS 1	3	50-400	0	2775	HI	Y	N	600	C	1200											
SHERON HARRIS 2	3	50-401	0	2775	HI	Y	N	600	C	1200											
FARLEY 1	3	50-348	49688	2652	HI	Y	N	600	H	1100	.775	547	1234	1436	1923	37	70	71	350	790	822
FARLEY 2	3	50-364	49689	2652	HI	Y	N	600	H	1100											
NORTH ANNA 1	3	50-338	49665	2775	HI	Y	N	600	H	1100											
NORTH ANNA 2	3	50-339	49666	2775	HI	Y	N	600	H	1100											
SURRY 1	3	50-280	49681	2441	HI	N	N	600	H	1100											
SURRY 2	3	50-281	49682	2441	HI	N	N	600	H	1100											
BEAVER VALLEY 1	3	50-334	49695	2652	HI	Y	N	600	H	1100											
BEAVER VALLEY 2	3	50-412	0	2652	HI	Y	N	600	H	1100	.775	547	1147	1436	1923	30	70	71	270	790	822
H.B. ROBINSON	3	50-261	49690	2292	LO	Y	N	600	H	1100	.775	547	1247	1749	1835	31	82	80	309	1050	902
TURKEY POINT 3	3	50-250	49679	2200	LO	Y	N	600	H	1100											
TURKEY POINT 4	3	50-251	49680	2200	LO	Y	N	600	H	1100											
PRAIRIE ISLAND 1	2	50-282	49663	1650	IN	N	N	700	H	1100	.775	547	1369	1797	1751	39	85	96	389	1100	1047
PRAIRIE ISLAND 2	2	50-306	49664	1650	IN	N	N	700	H	1100											
KEWAUNEE	2	50-305	49671	1650	IN	N	N	700	H	1100	.775	547	1253	1797	1751	38	85	96	361	1100	1047
R.E. GINNA	2	50-244	49677	1520	LO	Y	N	700	H	1100	.775	547	1199	1181	1528	29	74	72	305	640	800
POINT BEACH 1	2	50-266	49656	1520	LO	Y	N	700	H	1100											
POINT BEACH 2	2	50-301	49657	1520	LO	Y	N	700	H	1100											

APPENDIX A

PUMP-OPERATION CRITERIA THAT CAN RESULT IN RCP TRIP DURING TRANSIENTS AND ACCIDENTS

A. The NRC staff has concluded that if sufficient time exists, manual action is acceptable for tripping the RCPs following a LOCA provided certain conditions are satisfied.

B. Potential problem areas should be considered in developing RCP-trip setpoints and methods.

1. Tripping RCPs causes loss of pressurizer sprays.

   a. This produces a need to use PORVs in some plants to control primary pressure.

   b. PORVs have frequently failed to close.

   c. Despite testing, PORV operational reliability has not improved significantly.

2. Tripping RCPs tends to produce a stagnant region of hot coolant in the reactor-vessel upper elevations.

   a. Hot stagnant coolant has flashed and partially voided the upper vessel region during depressurization or cooldown operational events.

   b. Operators are not completely familiar with the significance of an upper-head steam bubble.

   c. Operators have difficulty controlling coolant conditions to avoid or control flashing.

   d. Operators may take precipitous actions when a steam bubble exists.

3. After tripping the RCPs, decay-heat removal by natural circulation is required. This procedure is used less frequently than controlling with the RCPs and it places more demand on the operators to control the primary-system conditions.

C. Consider the following guidelines in developing RCP-trip setpoints.

1. Demonstrate and justify that proposed RCP-trip setpoints are adequate for small-break LOCAs but will not cause RCP trip for other non-LOCA transients and accidents such as SGTRs.

a. Assure that RCP trip will occur for all primary-coolant losses in which RCP trip is considered necessary.

b. Assure that RCP trip will not occur for SGTRs up to and including the design-basis SGTR.

c. Assure that RCP trip will not occur for other non-LOCA transients where it is not considered necessary.

d. Perform safety analyses to prove that a, b, and c above are achieved.

e. Consider using partial or staggered RCP-trip schemes.

f. Assure that training and procedures provide direction for use of individual steam generators with and without operating RCPs.

g. Assure that symptoms and signals differentiate between LOCAs and other transients.

h. (Westinghouse plant specific) RCP trip is expected to occur for the design-basis SGTR for some Westinghouse plants that have SI pumps with lower shutoff heads. The exact rupture size above which RCP trip would be required has not been determined.

   (1) NRC informed Westinghouse that RCPs should not be tripped for SGTR events such as that which occurred at Ginna, which was essentially equivalent to a design-basis SGTR.

   (2) NRC informed Westinghouse that methods should be examined for either improving the RCP-trip setpoints or modifying the plants so that RCPs need not be tripped for design-basis SGTRs.

   (3) Restart permission was granted for Ginna with the requirement that supplementary guidelines be developed for RCP trip to assure RCPs would not be tripped for the design-basis SGTR.

   (4) NRC agrees with Westinghouse that in the long term using the reactor-vessel-liquid-inventory system to help determine when to do a RCP trip will increase the probability of maintaining RCP operation during non-LOCAs.

i. (Westinghouse plant specific) NRC has concluded that recent information by Westinghouse about wide-range pressure uncertainty indicates that analyses confirming Westinghouse's conclusions about RCP trip setpoints for high-head-SI plants are probably necessary.

2. Exclude extended RCP operation in a voided system where pump head is more than 10% degraded unless analyses or tests can justify pump and pump-seal integrity when operating in voided systems.

3. Avoid challenges to the PORVs where possible.

   a. If setpoints lead to RCP trip even though it is neither required nor desirable for transients or accidents with offsite power available, assure that challenges to the PORVs are avoided that would normally be handled by using pressurizer sprays.

   b. Challenges to PORVs could be eliminated by using heated auxiliary pressurizer sprays from a source other than the RCP discharge.

   c. If submittal recommends use of PORVs to depressurize, then licensees need to develop a program for upgrading the PORVs' operational reliability.

4. Establish guidelines and procedures for cases where RCP trip can lead to hot, stagnant fluid regions at primary-system high points.

   a. Describe symptoms of primary-system voiding caused by flashing of hot, stagnant fluid regions including effects on the pressurizer.

   b. Specify guidance for detecting, managing and removing the voids.

   c. Train operators concerning the significance of primary-system voids for both non-LOCA and LOCA conditions.

5. Assure that containment isolation will not cause problems if it occurs for non-LOCA transients and accidents.

   a. Demonstrate that, if water services needed for RCP operation are terminated, they can be restored fast enough once a non-LOCA situation is confirmed to prevent seal damage or failure.

   b. Confirm that containment isolation with continued pump operation will not lead to seal or pump damage or failure.

6. RCP-trip decision parameters should provide unambiguous indicators that a LOCA has occurred and the NRC-required inadequate-core-cooling instrumentation should be used where useful in indicating the need for a RCP trip.

7. NRC recommends that the licensee use event trees to systematically evaluate their set points to minimize the potential for undesirable consequences because of a misdiagnosed event.

a. Evaluate set points for events with RCP trip when it is preferable the RCPs remain operational.

b. Evaluate set points for events where early RCP trip does not occur and a delayed trip may lead to undesirable consequences.

D. NRC's guidance for justification of manual RCP trip in the licensee submittals is summarized in this section. This guidance had two purposes. It was intended to assist plants that can and should rely on manual trip to justify it, and it was also intended to help identify those few plants that may not be able to rely on manual trip.

1. Analyses should demonstrate that the limits set forth in 10 CFR 50.46 are not exceeded for the limiting small-break size and location using the RCP-trip set points developed with the guidance of part C above.

   a. Assume manual RCP trip does not occur earlier than 2 minutes after the RCP-trip set point is reached.

   b. Include allowances for instrument error.

   c. Generic analyses are acceptable if they are shown to bound the plant-specific evaluations.

2. Determine the time available to the operator to trip the RCPs for the limiting cases if manual RCP trip is proposed.

   a. Perform the analysis for the limiting small-break size and location identified in D.1 above.

   b. Use the most probable best-estimate analysis to determine the time available to trip the RCPs following the time when the RCP-trip signal occurs.

   c. Most probable plant conditions should be identified and justified by each licensee.

   d. NRC will accept conservative estimates in the absence of justifiable most probable plant conditions.

   e. Justify that the time available to trip the RCPs is acceptable if it is less than the Draft ANSI Standard N660.

      (1) Include an evaluation of operating experience data.

      (2) Address the consequences if RCP trip is delayed beyond this time.

      (3) Develop contingency procedures and make them available for the operator to use in case the RCPs are not tripped in the preferred time frame.

(4) No justification is required if the time available to trip the RCPs exceeds the Draft ANSI Standard N660.

E. Assure that good engineering practice has been used for the following areas.

1. Establish the quality level for the instrumentation that will signal the need for RCP trip.

   a. Identify the basis for the sensing-instruments' design features chosen.

   b. Identify the basis for the sensing-instruments' degree of redundancy.

   c. Licensees can take credit for all equipment available to the operators that they have sufficient confidence in its operability during the expected conditions.

2. Ensure that emergency operating procedures exist for the timely restart of the RCPs when conditions warrant.

3. Instruct operators in their responsibility for tripping RCPs for small-break LOCAs including priorities for actions after the engineered safety features actuation occurs.

APPENDIX B

PUMP-OPERATION CRITERIA THAT WILL NOT RESULT IN RCP TRIP DURING TRANSIENTS AND ACCIDENTS

Consider the following guidelines if the submittal concludes that keeping the RCPs running is both the preferred and safest method of pump operation for small-break LOCAs and other transients and accidents.

A. Evaluate inventory loss.

1. Complete evaluation of LOFT Test L3-6 through the ECCS recovery phase.

2. Evaluate all modeling differences expected between LOFT and a PWR analysis.

B. Evaluate pump integrity.

1. Justify how the pump-seal and pump structural integrity will be assured during extended two-phase flow performance.

2. Include the consequences of pump and/or pump-seal failure in the analyses if their integrity cannot be assured.

3. Include one of the following if continuous RCP operation is expected even with a containment isolation signal.

a. Evaluate the capability to continue RCP operation without essential water services.

b. Evaluate the capability to rapidly restore essential water services.

4. Evaluate the RCP's capability to operate in the accident environment.

5. Evaluate the consequences of RCP failure at any time during the accident if continuous operation in the accident environment cannot be assured.

C. Ensure acceptability of results.

1. Analyses should demonstrate that the 10 CFR 50.46 ECCS acceptance criteria are met with a model in compliance with Appendix K to 10 CFR Part 50.

2. Assume continuous pump operation and also RCP trip at various times if continuous pump operation cannot be assured.

3. NRC will consider a request for an exemption to 10 CFR 50.46 requirements if analyses indicate compliance cannot be achieved.

a. Submittal concludes that compliance with 10 CFR 50.46 would require operating the plant in a less safe condition. This

needs to be supported with a risk/benefit analysis that can take credit for all equipment expected to remain operational during the accident.

b. Submittal concludes that design modifications would not be cost-effective to implement from a safety standpoint.