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SYSTEMS DIVISION

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U. S. Nuclear Regulatory Commission
Division of Project Management
Office of Nuclear Reactor Regulation
Washington, D. C. 10555

Attention: Mr. Denwood F. Ross
Acting Director, Division of Project Management

Gentlemen:

SUBJECT: ADDITIONAL INFORMATION ON LOSS OF FEEDWATER TRANSIENT ANALYSIS

References:
1) Letter, D. F. Ross to T. D. Keenan "Additional Information Required to Evaluate NEDO-24708," September 28, 1979
2) "Additional information required for NRC Staff Generic Report on Boiling Water Reactors," NEDO-24708, August 1979

Reference 1 requested that additional information be supplied concerning the loss of feedwater transient analyses presented in Reference 2. Accordingly, enclosed on behalf of the BWR Owners Group are 60 copies of the responses to the questions presented in reference 1.

If you have any questions or comments, please contact S. J. Stark, (408) 925-1822, of my staff.

Very truly yours,

S. J. Stark for R. H. Buchholz

R. H. Buchholz, Manager
BWR Systems Licensing
Safety and Licensing Operation

RHB:at/101I

Enclosures

cc: L. S. Gifford (Bethesda)
BWR Owners Group
P. W. Marriott
T. D. Keenan

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ATTACHMENT

ADDITIONAL INFORMATION ON LOSS OF FEEDWATER TRANSIENT ANALYSES

Question 1. Provide a detailed discussion of how the SAFE code simulates a transient such as loss of feedwater (LOFW). Describe all input parameters and output parameters used for determining the Sequence-of-Events tables in NEDO-24708. Provide a comparison of the results of a SAFE code simulation with the normal transient code (REDY/ODYN) for each reactor class. Describe all modifying assumptions made when using the SAFE code to simulate transients.

Response 1. The LOFW analyses presented in NEDO-24708 were based on the following assumptions: (a) May-Witt (BWR/2-4) or ANS+20% (BWR/5) for decay heat, (b) the Appendix K model for the core heat transfer coefficient (h=4 following core spray initiation and h as a function of the void fraction below the two-phase level) and number of ADS valves operable, and (c) instantaneous feedwater flow shutoff to zero. These assumptions were conservative but would not yield misleading predictions of the system performance. In the development of the LOFW guidelines, more nominal analysis will be done and presented.

A detailed discussion of the use of SAFE code in the simulation of LOFW transients will be presented in the analyses to support the LOF guidelines. This will include a discussion of major input parameters and output parameters used for determining the Sequence-of-Events tables, a comparison of the results of a SAFE code simulation with a REDY code simulation for each reactor class (BWR/2-5), and a description and justification of major modifying assumptions made when using the SAFE code to simulate more nominal LOFW transients.

Question 2. Provide details on how BWR/1 transients were derived from the BWR/2 analyses.

Response 2. The details on how the BWR/1 LOFW sequence of events were derived from BWR/2 analyses will be presented in the analyses to support the LOFW guidelines.

Question 3. Provide a complete set of curves for the BWR/4 LOFW analyses. These should include: vessel level, vessel pressure, steam and feedwater flow, safety relief valve flow, ECC flows, steam line pressure, peak fuel temperature, bypass valve flow, neutron flux. For other reactor classes provide vessel pressure, vessel level, SRV flow, ECCS flows, steam flow, feedwater flow.

Response 3. A complete set of curves for the BWR/2-5 LOFW analyses will be presented in the analyses to support the LOFW guidelines. These will include: feedwater flow, steam line flow, core flow, reactor power, total heat to coolant, vessel pressure, safety/relief valve flow, ADS flow, vessel level (inside shroud, outside shroud), fuel temperature, high pressure auxiliary system flow, ECCS flow.

Question 4. Identify the representative plant in each reactor class and the rationale for selection. Describe how representative plants provide plant specific transient response when systems characteristics of plants differ within each reactor class.

Response 4. The representative plant selected for LOFW analysis in each reactor class (BWR/2-5) was as follows:

BWR/2	(EC/FWCI):	Nine Mile Point
BWR/3	(EC/FWCI or HPCI):	Millstone
BWR/3,4	(RCIC/HPCI):	Browns Ferry
BWR/5	(RCIC/HPCS):	Zimmer

The reactor class was specified such that all plants within a given class would have essentially the same LOFW sequence of events, which was determined mainly by the type and capacity of high pressure systems available. Therefore, any plant within a given reactor class could be used for analysis. The above plants were selected mostly for convenience.

For BWR/1, the LOFW sequence of events was derived from BWR/2 analyses and consideration of plant-unique parameters. Since there are plant-unique differences in BWR/1 reactor response, depending on whether a particular reactor will isolate automatically during a high-power LOFW event with CRD available, the sequence of events in NEDC-24708 was intended to cover both cases (with the non-isolation case covered in parenthesis).

Question 5. For the BWR/1 reactor with LOFW and no control rod drive (CRD) flow, show that the operator has one hour to manually isolate the reactor before core uncovery.

Response 5. For the BWR/1 reactor which does not isolate automatically during a high-power LOFW event with CRD available, i.e., the low-level isolation setpoint is considerably below the low-level scram setpoint, the liquid inventory between the two level setpoints is approximately 30,000 lb. Based on this and the ANS-5 decay heat, it was calculated that it would take about one hour (after scram) for the water level to drop to the isolation setpoint due to loss of feedwater/CRD and boiling-off. Therefore, the operator has one hour to manually isolate the reactor before it isolates automatically.

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Question 6. It is not apparent that additional failure in shutdown methods would not aggravate or change the course of a simulated transient as stated in NEDO-24708. Clarify.

Response 6. All paragraphs in NEDO-24708 with the heading "The Effect of Single Failures and Operator Errors" will be deleted from the writeup when the analyses to support the guidelines are submitted. Any consideration of additional failure in shutdown methods will be addressed in the Shutdown Operator Guidelines instead.

Question 7. For BWR/1 with no emergency condenser (EC) or CRD flow, provide the system response when the SRV recloses instead of remaining open. The pressure will rise again to the SRV set point and continue this cycling at high pressure while inventory is being depleted. If manual action is required, provide the instrumentation available to alert the operator and what actions are required to maintain acceptable core inventory.

Response 7. The system response for BWR/1 with no EC and CRD flow and with safety valve cycling will be presented in the analyses to support the LOFW guidelines. The manual action required to maintain acceptable core inventory will also be provided.

Question 8. For the BWR classes where the SRV cycle before decay heat is removed by ECCS, what happens to the vessel inventory? Provide plots of level, pressure, ECCS, and SRV flows.

Response 8. See response 3.

Question 9. It appears that a stuck open relief valve (SORV) combined with a LOFW and failure of high pressure systems is not as severe as a properly operating SRV or one that is partially stuck open. In determining the course of a LOFW transient a sensitivity study should be performed for determining operator action times for event recognition and proper mitigation.

Response 9. The sequence of events and operator action times for event recognition and proper mitigation for the case of LOFW with failure of high pressure systems and a properly operating SRV will be presented in the analyses to support the LOFW guidelines. (See Response 7, also.) Studies will be conducted for a stuck-open safety relief valve and a properly operating valve. The system response and operator action times for these two cases will bracket the results of the case where there is a partially stuck open SRV.

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Question 10. Justify the assumptions used in the analyses to show operator action times as provided in the sequence of events. For example, justify the selection used for decay heat which varied for reactor class. How sensitive is the analysis to your assumptions?

Response 10. See Response 1.

Question 11. It appears that operationally i[illegible] de[illegible]red to manually restart a failed high pressure system prior to using the automatic depressurization system (ADS) for the low pressure (LPCI/LPCS) ECCS. However, the core inventory recovery is faster with [illegible] no high pressure ECCS) and LPCI/LPCS. What will the guidelines suggest to the operator?

Response 11. This response will be provided in the LOFW operator guidelines.

Question 12. Supply curves to show the differences in SRV opening times and level recovery times for BWR/4 and BWR/5 reactors.

Response 12. See Response 3.

Question 13. Provide the analyses and sequence of events for the LOFW coupled with a stuck open SRV and the following: loss of offsite power, loss of all A-C power, and loss of one train of D-C power with loss of offsite power. Provide the following time-dependent variables: SRV flow, vessel pressure, ECCS flows, vessel water level, and fuel temperatures. The initial conditions assumed in the analyses should be provided and the time at which stable conditions are reached. If core uncovery results, provide the basis for assessing core damage.

Response 13. The analyses and sequence of events for the LOFW transient coupled with an SORV and loss of offsite power and diesels will be presented in the analyses to support the LOF guidelines. A complete set of curves (as detailed in Response 3) for the BWR/2-5 analyses will be provided.

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