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CRUTCHFIELD,D. Operating Reactors Branch 5

SUBJECT: Forwards responses to NRC 801031 ltr re clarification of TMI action plan requirements.Util intends to comply w/NRC requirements & implementation dates.Execptions are discussed in encl.Deviations will be communicated.

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JOHN E. MAIER
VICE PRESIDENT

TELEPHONE
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December 15, 1980

Director of Nuclear Reactor Regulation
Attention: Mr. Dennis M. Crutchfield, Chief
Operating Reactors Branch No. 5
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Clarification of TMI Action Plan Requirements (NUREG 0737)
R. E. Ginna Nuclear Power Plant
Docket No. 50-244

Dear Mr. Crutchfield:

Darrel Eisenhut's October 31, 1980 letter provided clarification of TMI Action Plan Requirements and placed in one document all TMI-related items approved for implementation by the Commission at this time. The letter also provided implementation dates for each item and requested that we confirm that the implementation dates would be met.

In general, we intend to comply with the requirements and implementation dates as set forth in Mr. Eisenhut's letter. Exceptions to the requirements or dates are discussed in Attachment A. Because we have had relatively little time to analyze some of the requirements of Mr. Eisenhut's letter, we may inform you of additional deviations at a later date. Our commitment to the dates in Mr. Eisenhut's October 31, 1980 letter supersedes all previous commitments which indicated Action Plan item completion dates earlier than those in NUREG-0737.

Very truly yours,

John E. Maier

John E. Maier

Subscribed and sworn to me
on this 15th day of December, 1980

Gary L. Reiss

GARY L. REISS
NOTARY PUBLIC, State of N. Y. Monroe Co.
My Commission Expires March 30, 19.81.

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ATTACHMENT A

Rochester Gas and Electric Corporation

Response to Clarification of TMI Action Plan Requirements

December 15, 1980

The following comments and commitments are stated in response to NUREG 0737, Clarification of TMI Action Plan Requirements. Item numbers correspond to those in the NUREG. Only those items for which RG&E has comments or proposes deviations from the schedule or requirements are listed.

1.A.1.1 Shift Technical Advisor

RG&E's most recent response concerning Shift Technical Advisors is contained in a letter dated August 5, 1980 from L. D. White, Jr. to Mr. Dennis M. Crutchfield, USNRC. We have implemented a program to provide trained STAs by January 1, 1981, however, because the program was redirected in response to a letter dated July 7, 1980 from Mr. Crutchfield, the individuals will not have one full year of training, a potential requirement implied in a letter dated September 13, 1979 from Mr. Darrell Eisenhut. A complete description of our STA program will be provided by January 1, 1981.

1.A.1.3 Shift Manning

By letter dated October 13, 1980 from L. D. White, Jr. to Mr. Dennis M. Crutchfield, USNRC, RG&E responded to shift staffing criteria and guidelines for scheduling overtime for licensed operators. The commitments provided in that letter, and proposed alternatives to some of the Staff overtime guidelines, remain unchanged. By January 1, 1981 we will propose a policy to limit overtime work of people in addition to licensed operators who perform safety related work.

I.C.1 Guidance for the Evaluation and Development of Procedures for Transients and Accidents

The Westinghouse Owners Group will submit by January 1, 1981, a detailed description of our program to comply with the requirements of Item I.C.1. The program will identify previous Owners Group submittals to the NRC, which we believe will comprise the bulk of the response. Additional effort required to obtain full compliance with this item (with proposed schedules for completion) will also be identified, as discussed with the NRC on November 12, 1980.

I.C.6 Guidance on Procedures for Verifying Correct Performance of Operating Activities

We intend to meet the intent of the requirements of this position as it is applied to safety related equipment and systems. The Staff recognized in NUREG 0737 that licensed operators are qualified to perform return-to-service verification of equipment important to safety and is investigating the level of qualification for other operators to perform these functions. We intend to use individuals knowledgeable of the systems involved but not necessarily licensed operators.

I.D.1 Several of the Action Plan items now require a human engineering review. In some cases equipment has already been procured and installed to meet an NRC deadline. We do not intend to redesign or reinstall equipment except as may be required after review of the forthcoming NRC guidelines (NUREG 0700).

II.B.1 Reactor Coolant System Vents

Requirements have been changed since we designed our system and submitted it for NRC review nearly one year ago. A good faith effort was made to meet the NRC's earlier deadline for installation of January 1, 1981 and our system is now operational. Good engineering judgment was used in the design of the system and we believe we meet all of the requirements, including those recently imposed. Because the system is installed we do not plan to make modifications to it. Tentative approval was given to our system in a letter dated February 29, 1980 from Dennis L. Ziemann, USNRC, to L. D. White, Jr. contingent upon our verification that the system met certain requirements. That verification was provided by a letter dated June 2, 1980 from Mr. White to Mr. Ziemann. Please confirm that our installed system is acceptable.

II.B.2 Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems Which May Be Used in Post-Accident Operations

Modifications to the plant necessary for post-accident operations and committed to be performed in our letter dated December 28, 1979 from L. D. White, Jr. to Mr. Dennis Ziemann, USNRC will be completed by the required date of January 1, 1982. Our December 28, 1979 response also said we would investigate a modification to the plant radwaste equipment procedures and controls. We have performed that investigation and determined it is prudent to duplicate some controls on the radwaste panel at a remote location. As engineering has progressed and cost estimates have been developed, it is apparent that the cost of moving some controls is nearly as great as the cost of total replacement of the existing radwaste control panel with a computer-based digital control system. A major advantage to implementing the computer-based system would be that control of the Radwaste System could be from several remote locations. This would eliminate the need for an operator to be stationed in the area of the existing panel, even during normal operation, and thus eliminate radiation exposure. The computer-based system would therefore contribute to reducing the radiation exposure of the plant operators and contribute to our ALARA effort. Since the design of a computer-based control system

involves current state-of-the-art technology and system development, the total estimated project length is longer than that of the remote duplication effort. For this reason RG&E will not be able to complete this modification by January 1, 1982. Our present schedule, however, projects completion by end of the Spring 1982 Refueling Outage.

The design of our shielding modifications is ongoing, however not complete. Thus all design details will not be available by January 1, 1981.

II.D.1 Performance Testing of BWR and PWR Relief and Safety Valves

As a sponsor of the EPRI PWR Safety and Relief Valve Test Program, Rochester Gas and Electric Corporation intends to comply with the requirements of NUREG 0578, Item 2.1.2. By letter dated December 15, 1980, R. C. Youngdahl of Consumers Power Company has provided the current PWR Utilities' positions on NUREG 0737, Item II.D.1 clarifications. Briefly those positions are:

A. Safety and Relief Valves and Piping - the EPRI "Program Plan for Performance Testing of PWR Safety and Relief Valves", Revision 1, dated July 1, 1980, does provide a program that satisfies the NRC requirements. Discussion with the NRC staff and their consultants is resolving specific detailed issues.

B. Block Valves - The EPRI Program has not formally included the testing of block valves. However, a small number of block valves have been tested at the Marshall Steam Station Test Facility. The PWR Utilities and EPRI cannot provide a detailed block valve test program until results of the Wyle and CE relief valve tests are available. Therefore, a block valve test program will not be provided before July, 1981. The PWR Utilities and EPRI believe that the proper operation of the TMI-2 and Crystal River block valves and other operational experience, plus knowledge of the Marshall tests, support a less hurried and more rational approach to block valve testing.

C. ATWS Testing - PWR Utilities will not support additional efforts for ATWS valve testing until regulatory issues are resolved. The major safety and relief valve test facility (CE) is nearing completion and some measures were taken to provide additional test capability beyond the current program requirements. Results from the current

program are likely to provide most of the information necessary to address ATWS events (i.e., relief capability at high pressures).

II.E.4.2 Containment Isolation Dependability

The purge and vent system at Ginna consists of four 48 inch isolation valves. The Staff's interim position on containment purging (now called Position 6) was implemented by our December 14, 1979 and May 29, 1980 letters. During a recent review of Position 6, it was postulated that two 6 inch valves on our containment depressurization line may be interpreted as falling under Position 6 requirements. These valves are not used for containment purge and vent operations but are used periodically to equalize pressure between inside and outside containment. If these valves are not allowed to be opened during normal operation, unacceptable containment pressures may result. We are reviewing the operability of these valves under accident conditions. In addition we are reviewing the effects on the containment and the containment isolation pressure setpoint if this line is not used periodically to minimize containment pressure fluctuation. We concur with the basic requirement that the containment pressure setpoint that initiates containment isolation of nonessential penetrations should be the minimum setpoint compatible with normal operating conditions. However, pending additional evaluation, we cannot concur with the clarification statement that 1 PSI above normal operation pressure is an appropriate minimum pressure setpoint. This value may not be sufficient to account for instrumentation drift and accuracy as well as abnormal operation scenarios which may increase containment pressure above normal but for which isolation of nonessential penetrations is not desirable. It should be noted that the same pressure setpoint that actuates containment isolation also actuates safety injection and trips the reactor. By January 1, 1981 we will address how the intent of Position 6 will be applied to the depressurization valves.

II.F.1.1 Noble Gas Effluent Monitor

The possible release paths for noble gases from the plant secondary system have been identified as the main steam safety and relief valves, turbine driven auxiliary feedwater pump (TAFWP) exhaust, and main condenser air ejector. An adjacent steam line monitor having a detection range of 10^{-1} - 10^{+3} µCi/cc is proposed on each main steam line upstream of the safety valves and TAFWP supply lines. Any releases from the safeties, TAFWP exhaust, or condenser air ejector can then be quantified by integrating the flow rate times the steam activity concentration with respect to time.

Condenser exhaust gases are diluted by a factor of approximately 100. A low range detector (10^{-6} - 10^{-3} μCi/cc) presently monitors the condenser air ejector exhaust as a means of early detection of primary to secondary leaks. We propose to increase the range of this detector to overlap with the 10^{-1} - 10^{+3} μCi/cc range of the steam line monitors, thus providing a means for following a continuous increase in steam activity from 10^{-6} to 10^{+3} μCi/cc.

A low range detector on the TAFWP exhaust is believed unnecessary since the air ejector monitor will be viewing steam activity levels under normal operations. The TAFWP is not normally in use during power operation when a low range monitor on the steam exhaust might be used to detect steam generator tube leaks. Tube leaks will normally be detected by the air ejector monitor.

A detailed system description including detector models, data recording methods, and calculational procedures will be submitted by February 1, 1981.

Clarification 4.a.(iv) requires a capability to obtain readings at least every 15 minutes during and following an accident. Table II.F.1-1 requires the display to be continuous and recording as equivalent Xe-133. These requirements are inconsistent, however, we assume that the Eberline Sping-4's purchased and now on site for monitoring the containment vent and the plant vent are acceptable. We previously documented our intention to use these monitors on December 28, 1979. The Sping-4's are capable of automatically logging activity levels every ten minutes when in alarm or at any time when manually requested by the operator.

II.F.1.3 Containment High Range Radiation Monitor

A Victoreen Model 875 High Range Containment Area Monitor System has been purchased for installation by January 1, 1982. The system is currently being qualified to IEEE-323 and Regulatory Guides 1.97 and 1.89, with test reports expected to be completed by March 1981. Until those tests are complete, however, we cannot commit that the installed system will meet all of the NRC requirements.

II.F.1.4 Containment Pressure Monitor

The Staff position presently calls for "continuous recording" of containment pressure; it is felt that this would result in a waste of paper and unnecessary wear on the recorder mechanism. A system is proposed, however, that will start recording whenever a safety injection or containment isolation signal is present.

This proposed system will provide adequate recording of signals.

II.F.1.5 Containment Water Level Monitor

The narrow range containment water level instrument (Sump A) will provide useful information only for leaks or small fluid line breaks. It will provide no useful information for most LOCA break sizes. Thus, the narrow range instrumentation should be designed and qualified for those postulated accident conditions for which operability of the instrumentation can provide meaningful indication. Regulatory Guide 1.89 is an appropriate reference for this instrumentation, however, the qualification envelope is based upon the environment anticipated for small leaks and breaks.

The wide range containment water level instrument (Sump B) is designed to monitor water depths corresponding to at least 500,000 gallons. Because we could not provide documentation of the qualification of the existing instrumentation, we committed in our Environmental Qualification of Electrical Equipment letter dated October 31, 1980 from John E. Maier to Mr. Darrell G. Eisenhut to replace this instrumentation. It will be replaced by June 30, 1982.

II.F.2 Instrumentation for Detection of Inadequate Core Cooling

A letter dated July 2, 1980 from L. D. White, Jr. to Mr. Dennis M. Crutchfield provided RG&E's position concerning installation of additional instrumentation for the purpose of determining reactor vessel water level. Our position is unchanged since that time. In essence, we do not believe that a reliable, easy-to-interpret, unambiguous indication of vessel level has been demonstrated to exist in any of the devices available for installation at this time. We will continue to encourage and support the development of such a device, however, and when one has been successfully demonstrated to function properly over the range of conditions for which it is intended to operate, and shown to provide useful information to the operator, we will install additional instrumentation. Until that time we do not intend to install a so-called water level device. Additional information concerning our position is contained in our July 2 letter. No response to our letter has been received. Information received at a recent NRC sponsored seminar on water level devices in Idaho Falls reinforces our belief that much testing and confirmation work must be completed before water level devices should be installed in operating reactors.

Since the TMI accident, improved operator training, revised operating procedures based upon extensive new analyses, and installed subcooling meters have provided additional safety margin for adequate core cooling. Plant procedures for recognizing and mitigating inadequate core cooling coupled with existing instrumentation provide for all the mitigating actions available to the plant operator. If the core is determined to be inadequately cooled, knowledge of the specific water level (if something other than froth exists under transient conditions) will not affect the actions to be taken by the operator.

Therefore, continued operation of the R. E. Ginna Nuclear Power Plant with existing instrumentation and without the addition of a reactor vessel water level device will not adversely affect public health and safety.

II.K.2.13 Thermal Mechanical Report -- Effect of High-Pressure Injection on Vessel Integrity for Small-Break Loss-of-Coolant Accident with No Auxiliary Feedwater

To completely address the NRC requirements of detailed analysis of the thermal-mechanical conditions in the reactor vessel during recovery from small breaks with an extended loss of all feedwater, a program will be completed and documented to the NRC by the Westinghouse Owners Group by January 1, 1982. This program will consist of analysis for generic Westinghouse PWR plant groupings.

Following completion of this generic program, additional plant specific analyses, if required, will be provided. A schedule for the plant specific analysis will be determined based on the results of the generic analysis.

II.K.2.17 Potential for Voiding in the Reactor Coolant Systems during Transient

The Westinghouse Owners Group is currently addressing the potential for void formation in the Reactor Coolant System (RCS) during natural circulation cooldown conditions, as described in W Letter NS-TMA-2298 (T. M. Anderson, W. to P. S. Check, NRC). We believe the results of this effort will fully address the NRC requirement for analysis to determine the potential for voiding in the RCS during anticipated transients. A report describing the results of this effort will be provided to the NRC before January 1, 1982.

II.K.2.19 Sequential Auxiliary Feedwater Flow Analysis

The Transient Analysis Code, LOFTRAN, and the present small break evaluations analysis code, WFLASH, have both undergone benchmarking against plant information or experimental test facilities. These codes under appropriate conditions have also been compared with each other. The Westinghouse Owners Group will provide on a schedule consistent with the requirement of Task II.K.2.19, a report addressing the benchmarking of these codes.

II.K.3.1 Installation and Testing of Automatic Power-Operated Relief Valve Isolation System

And

II.K.3.2 Report on Overall Safety Effect of Power-Operated Relief Valve Isolation System

The Westinghouse Owners Group is in the process of developing a report (including historical valve failure rate data and documentation of actions taken since the TMI-2 event to decrease the probability of a stuck-open PORV) to address the NRC concerns of Item II.K.3.2. However, due to the time-consuming processing of data gathering, breakdown, and evaluation, this report is scheduled for submittal to the NRC on March 1, 1981. As required by the NRC, this report will be used to support a decision on the necessity of incorporating an automatic PORV isolation system as specified in Task Action Item II.K.3.1.

II.K.3.5 Automatic Trip of Reactor Coolant Pump During Loss of Coolant Accident

The Westinghouse Owners Group resolution of this issue has been to perform analyses using the Westinghouse Small Break Evaluation Model WFLASH to show ample time is available for the operator to trip the reactor coolant pumps following certain size small breaks (See WCAP-9584). In addition, the Owners Group is supporting a best estimate study using the NOTRUMP computer code to demonstrate that tripping the reactor coolant pump at the worst trip time after a small break will lead to acceptable results.

For both of these analysis efforts, the Westinghouse Owners Group is performing blind post-test predictions of LOFT experiment L3-6. The input data and model to be used with WFLASH on LOFT L3-6 has been submitted to the Staff on December 1, 1980 (NS-TMA-2348). The information to be used with NOTRUMP on LOFT L3-5 will be submitted prior to performance of the L3-6 test as stated in Westinghouse Owners Group letter OG-45 dated December 3, 1980.

The LOFT prediction from both models will be submitted to the Staff on February 15, 1981 given that the test is performed on schedule. The best estimate study is scheduled for completion by April 1, 1981.

Based on these studies, the Westinghouse Owners Group believes that resolution of this issue will be achieved without any design modifications. In the event that this is not the case, a schedule will be provided for potential modifications.

II.K.3.30 Revised Small Break LOCA Methods to Show Compliance with 10 CFR 50, Appendix K
and
II.K.3.31 Plant Specific Calculations to Show Compliance with 10 CFR 50.46.

Our response to these items was provided by a letter dated November 13, 1980 from John E. Maier to Mr. Dennis M. Crutchfield, USNRC.

III.A.2 Improving Licensee Emergency Preparedness - Long Term

At this time we believe we will be able to comply with the implementation schedule established for this item. However, we plan to comply with the requirement for a prompt notification system primarily with the installation of sirens. We do not yet have a commitment for supply of the sirens because field work necessary to establish sound levels, siren locations and the number of sirens required is not yet completed. If it becomes necessary to request an extension of the implementation date as this work proceeds, we will notify you promptly.

III.D.3.4 Control Room Habitability Requirements

The information requested in Attachment 1 to item III.D.3.4 will not be submitted January 1, 1981 for the reasons given in a letter dated November 24, 1980 from L. D. White, Jr. to Mr. Dennis M. Crutchfield, USNRC.

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