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CRUTCHFIELD,D. Operating Reactors Branch 5

SUBJECT: Forwards util response to 800507 ltr containing action items resulting from TMI action plan,including feedback procedures of operating experience to plant staff & worker radiation protection improvements.

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LEON D. WHITE, JR.
VICE PRESIDENT

TELEPHONE
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June 13, 1980

Director of Nuclear Reactor Regulation
Attention: Mr. Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Five Additional TMI Related Requirements
R. E. Ginna Nuclear Power Plant
Docket 50-244

Dear Mr. Crutchfield:

Darrel Eisenhut's letter dated May 7, 1980 contained licensee action items for operating reactors which are a result of the TMI Action Plan. The RG&E response to these items and the schedules proposed by the NRC Staff is contained in attachments A, B and C.

Sincerely yours,

L D White, Jr.

L. D. White, Jr.

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ATTACHMENT A

RG&E RESPONSE TO

ENCLOSURE 2

PROCEDURES FOR FEEDBACK OF

OPERATING EXPERIENCE TO PLANT STAFF

(Darrel Eisenhut letter dated May 7, 1980)

R. E. Ginna
June 11, 1980

1.C.5 Procedures for Feedback of Operating Experience To Plant Staff

Position

In accordance with Task Action Plan I.C.5, Procedures for Feedback of Operating Experience to Plant Staff, each licensee shall review its procedures and revise them as necessary to assure that operating information pertinent to plant safety originating both within and outside the utility organization is continually supplied to operators and other personnel and is incorporated into training and retraining programs. These procedure shall:

(1) Clearly identify organizational responsibilities for review of operating experience, the feedback of pertinent information to operators and other personnel and the incorporation of such information into training and retraining programs;

(2) Identify the administrative and technical review steps necessary in translating recommendations by the operating experience assessment group into plant actions (e.g., changes to procedures; operating orders);

(3) Identify the recipients of various categories of information from operating experience (e.g., Supervisory personnel, STA's, operators, maintenance personnel, H. P. technicians) or otherwise provide means through which such information can be readily related to the job functions of the recipients.

(4) Provide means to assure that affected personnel become aware of and understand information of sufficient importance that should not wait for emphasis through routine training and retraining programs;

(5) Assure that plant personnel do not routinely receive extraneous and unimportant information on operating experience in such volume that it would obscure priority information or otherwise detract from overall job performance and proficiency;

(6) Provide suitable checks to assure that conflicting or contradictory information is not conveyed to operators and other personnel until resolution is reached; and,

(7) Provide periodic internal audit to assure that the feedback program functions effectively at all levels.

Discussion

(omitted)

Clarification

Review of and modifications to procedures governing feedback of operating experience to plant staff shall be completed and the procedures put into effect on or before January 1, 1981.

Action (Class I)

(1) Licensee to implement actions and submit documentation of the method for staff review by scheduled dates.

RG&E Response:

1) The organizational responsibilities for review of operating experience, the feedback of pertinent information to operators and other personnel and the incorporation of such information into training and retraining programs is outlined in administrative procedure A-201, Ginna Station Administrative and Engineering Staff Responsibilities.

   The Technical Assistant Operational Assessment and the STA's maintain and upgrade safe plant operations by evaluation of applicable operating experience. The information reviewed includes LER's from similar design plants, NRC I&E bulletins, notices and circulars, Westinghouse circulars, and NPRD reports. The STA's also review Ginna Station LER's and in-house event reports. Recommendations are made for incorporation into plant operation.

   The Training Section reviews and acknowledges information received from the operations assessment group and incorporates information into their operator training and requalification programs.

2) Procedures exist to identify the administrative and technical review steps necessary to translate recommendations into plant operations. The Technical Assistant and the STA's meet on a regular basis to discuss information relevant to safe plant operations. Appropriate recommendations may be made to the Plant Superintendent, Operations Engineer, or Training Coordinator. These may be incorporated into plant operations or operator training programs by issuing instructions through the Operations Plan (outlined in procedure A-52.7) and through procedure change notification forms which are used for submitting new procedures or changing existing procedures as directed by procedure A-601, Plant Procedure Document Control.

3) Various information from operating experiences is routed to plant personnel through the Technical Assistant - Operational Assessment. Routine information is directed to the Plant Superintendent, Operations Engineer, Training Coordinator, and STA's. Specific information is directed to appropriate sections (e.g. maintenance, health physics). Section managers are responsible for distribution of applicable information to their personnel.

   Procedures will be developed for the feedback of pertinent information of operating experience. Recipients of various categories of information will be identified by these procedures.

4) Information of importance to safe plant operations requiring immediate emphasis is passed on to operational personnel through on-shift discussions originated by the STA's. These discussions are documented in the on-the-job training logs. Procedure A-102.14, Operator Requalification Program, addresses these requirements.

   Information requiring immediate implementation may be included in the Operations Plan as directed by Procedure A-52.7 or by changes to operating or emergency procedures.

5) Pertinent information related to operating experience is routed to plant staff by a limited number of individuals. Procedures for feedback of operating experience will be developed to incorporate means of assuring that plant personnel do not routinely receive extraneous and unimportant information.

6) Procedures for feedback of operating experience will be developed to provide suitable checks to ensure that conflicting and contradictory information is not conveyed to operational personnel until resolution is reached.

7) Periodic internal audits to assure that the feedback program functions effectively at all times will be incorporated into the Quality Assurance program and will be addressed by QA procedures.

We will comply with the staff's position as described in our response by January 1, 1981.

ATTACHMENT B

RG&E RESPONSE TO

<u>ENCLOSURE 3</u>

MEASURES TO MITIGATE SMALL BREAK

LOSS OF COOLANT ACCIDENTS AND

LOSS OF FEEDWATER ACCIDENTS

(Darrel Eisenhut's letter dated May 7, 1980)

R. E. Ginna
June 11, 1980

Installation and Testing of Automatic PORV Isolation System (II.K.3.1)

Position

(a) All PWR licensees should provide a system which uses the PORV block valve to protect against a small break LOCA. This system will automatically cause the block valve to close when the reactor coolant system pressure decays after the PORV has opened, to relieve excess pressure. An override feature should be incorporated. Justification should be provided to assure that failure of this system would not decrease overall safety by intensifying plant transients and accidents.

(b) Each licensee should perform a confirmatory test of the automatic block valve closure system installed in response to (a) above.

Clarification

Implementation of this action item has been modified in the forthcoming May 1980 version of NUREG-0660. The change delays implementation of this action item until after the studies specified in Action Item II.K.3.2 have been completed, if such studies confirm that the subject is necessary.

Discussion

NUREG-0565 (2.1.2.a)
NUREG-0611 (3.2.4.e, 3.2.4.f)
NUREG-0635 (3.2.4.a, 3.2.4.b)

Schedule

Design - July 1, 1981
Test - First refueling cycle

Action: (Class III)

(1) Licensee to document proposed changes for staff approval prior to implementation. Documentation to be submitted by scheduled date.

(2) Licensee to implement modifications and perform confirmatory test at the next refueling outage following staff approval of the design unless this outage is scheduled within six months of the approval date. In this event modifications will be completed during the following refueling outage.

RG&E Response:

Addition of automatic PORV block valve closure with manual override will increase the complexity of the existing safety system design, require additional operator actions

in both normal and emergency situations, and provide a questionable increase in system safety.

Emphasis has appropriately been placed on enhanced operator detection and recognition of the stuck open PORV, followed by manual block valve closure rather than on automatic block valve closure. This course of action clearly increases operational safety without introductng additional safety concerns.

Addition of automatic block valve closure on the other hand introduces several undesirable event sequences. Automatic closure of the PORV block valves could increase the challenge to the pressurizer safety valves during operational transients, thus increasing the probability of the small break LOCA via the safety valves. This occurrence would be more difficult to manage since there are no block valves in series with the code safety valves. The status of the override will have to be displayed as clearly as the PORV position itself, thus providing an operator distraction. The automatic closure of the block valves could defeat the Overpressure Protection System during cold, solid operation.

It is the RG&E position that analyses have demonstrated that the failure to isolate a stuck open PORV does not result in uncovering the core and that this in turn provides an adequate basis for relying on operator action to terminate the event. No modifications are presently being considered and unless the report requested in II.K.3.2 clearly indicates a need, none are planned for future implementation.

PWR Vendor Report on PORV Failure Reduction (II.K.3.2)

Position

(a) Each PWR vendor should submit a report for staff review documenting the various actions which have been taken to decrease the probability of a small break LOCA caused by a stuck-open PORV and show how they constitute sufficient improvements in reactor safety. This report should be submitted for staff review.

(b) Safety valve failure rate based on past history of the vendor designed operating plants should be included in the report submitted in response to (a) above.

Clarification

In addition to modifications already implemented on PORVs, the report specified above should include consideration of the automatic PORV isolation system identified in Action Item II.K.3.1. This item is applicable to PWRs.

Discussion

NUREG-0565 (2.1.2.d)
NUREG-0611 (3.2.4.g, 3.2.4.i)
NUREG-0635 (3.2.4.c)

Schedule

January 1, 1981

Action: (Class I)

Licensee to provide the report to the staff by scheduled date.

RG&E Response:

Westinghouse has been contacted about providing the requested report. Although arrangements have not been finalized we expect that the report can be completed by the end of the year.

Reporting Safety and Relief Valve Failures and Challenges (II.K.3.3)

Position

(a) Future failures of a relief valve to close should be reported promptly to the NRC.

(b) Future challenges to the relief valves should be documented in the annual report.

(c) Future failures of a safety valve to close should be reported promptly to the NRC.

(d) Future challenges to the safety valves should be documented in the annual report.

Clarification

This action item is applicable to all LWRs. Safety valve failure rate based on historical data is addressed in Action Item II.K.3.2.

Discussion

NUREG-0565 (2.1.2.c, 2.1.2.e)
NUREG-0611 (3.2.4.h, 3.2.4.j)
NUREG-0626 (F-2.5, F-3.5)
NUREG-0635 (3.2.4.d)

Schedule

April 1, 1980

Action: (Class I)

Licensee to provide annual report on SRV and RV failures and challenges as of April 1, 1980.

RG&E Response:

a) Future failures of a relief valve (on the primary or secondary systems) to close will be reported promptly to the NRC. Procedure O-9.3, NRC Immediate Notification will be revised to incorporate this requirement.

b) Future challenges to the relief valves on the primary and secondary systems as of April 1, 1980 will be documented and incorporated in the annual report submitted to the NRC for review.

c) Future failures of a safety valve (on the primary or secondary systems) to close will be reported promptly to the NRC (See response to (a) above).

d) Future challenges to the safety valves on the primary or secondary systems as of April 1, 1980 will be documented and incorporated in the annual report submitted to the NRC.

It should be noted that use of steam line power operated relief valves at Ginna Station is considered normal operation during plant startup and shutdown. These valves are required to maintain conditions necessary for secondary plant operation since Ginna does not have vacuum pumps. Operation of the steam line power operated valves under these conditions will not be considered as challenges and will not be documented in the annual report as noted in response (b) and (d) above.

We will comply with the staff's position as described in our response.

Automatic Trip of Reactor Coolant Pumps During LOCA (II.K.3.5)

Position

Tripping of the reactor coolant pumps in case of a LOCA is not an ideal solution. The licensees should consider other solutions to the small break LOCA problem (for example, an increase in safety injection flow rate). In the meantime, until a better solution is found, the reactor coolant pumps should be tripped automatically in case of a small break LOCA. The signals designated to initiate the pump trip should be carefully selected in order to differentiate betwen a small break LOCA and other events which do not require reactor coolant pump trip as discussed in NUREG-0623.

Clarification

Application to PWRs only. This action item has been revised in the May 1980 version of NUREG-0660 to provide for continued study of criteria for early reactor coolant system pump trip. Implementation, if any is required, will be delayed accordingly.

Discussion

NUREG-0565 (2.3.2.a)
NUREG-0611 (3.2.2.a)
NUREG-0635 (3.2.2.a)
NUREG-0623 (7.3)

Schedule

Study: January 1, 1981

Modify: January 1, 1982

Action: (Class I)

Licensee to provide results of evaluation of alternate solution to reactor coolant pump trips to staff by scheduled date.

RG&E Response:

The Westinghouse Owners Group analysis of delayed RCP trip during small break LOCAs is documented in WCAP-9584. This WCAP is the basis for the Westinghouse and Owners Group position that automatic RCP trip is not necessary for a Westinghouse PWR since sufficient time is available for manual tripping of the RCP. This philosophy has been incorporated in the Westinghouse emergency operating instructions which were reviewed and approved by the NRC Bulletins and Orders Task Force and subsequently incorporated in the Ginna emergency operating procedures. In addition, the Westinghouse criteria for RCP trip with RCS pressure below the shutoff head of SI pumps, provides

for continued RCP operation and therefore forced circulation and decreased reliance on operator action for non-LOCA events. The NRC has indicated that small break tests at the Semiscale and LOFT facilities will aid in NRC resolution of this issue. Therefore, we believe that it is not appropriate to take any additional actions on this issue until the NRC sponsored testing (in particular L3-5 and L3-6) is completed and the results evaluated.

Proportional Integral Derivative (PID) Controller Modification (II.K.3.9)

Position

The Westinghouse-recommended modification to the Proportional Integral Derivative (PID) controller should be implemented by affected licensees.

Clarification

This action item is applicable only to Westinghouse-designed PWRs.

Discussion

NUREG-0611 (3.2.4.a)

Schedule

July 1, 1980

Action: (Class II)

Licensee to implement actions and submit documentation of the method for staff review by schedule date.

RG&E Response:

The PID controller has been modified to comply with the Westinghouse recommendation. Documentation of the changes is available for inspection at the plant.

Proposed Anticipatory Trip Modification (II.K.3.10)

Position

The anticipatory trip modification proposed by some licensees to confine the range of use to high power levels should not be made until it has been shown on a plant-by-plant basis that the small break LOCA probability resulting from a stuck-open power-operated relief vave (PORV) is little affected by the modification.

Clarification

This action item is applicable only to Westinghouse-designed PWRs.

Discussion

NUREG-0611 (3.2.4.c)

Schedule

Plant by plant

Action: (Class III)

1) Licensee to document proposed change for staff approval prior to implementation. Documentation to be submitted as proposed by the licensee.

2) Licensee to implement modifications at the next refueling outage following staff approval of the design unless this outage is scheduled within six months of the approval date. In this event modifications will be completed during the following refueling outage.

RG&E Response:

RG&E has not proposed to modify the anticipatory trip at R. E. Ginna. The anticipatory trip function incorporated in the original plant design and license is described in Chapter 7 of the FSAR. Analyses of loss of electrical load transients are given in Chapter 14 of the FSAR.

Confirm Existence of Anticipatory Trip Upon Turbine Trip (II.K.3.12)

Position

Licensees with W-designed operating plants should confirm that their plants have an anticipatory reactor trip on turbine trip. The licensee of any plant where this trip is not present should provide a conceptual design and evaluation for the installation of this trip.

Clarification

This item is applicable to Westinghouse PWRs.

Discussion

NUREG-0611 (3.2.4.a)

Schedule

July 1, 1980

Action: (Class III)

(1) Licensee to document proposed changes for staff approval prior to implementation. Documentation to be submitted by scheduled date.

(2) Licensee to implement modifications at the next refueling outage following staff approval of the design unless this outage is scheduled within six months of the approval date. In this event modifications will be completed during the following refueling outage.

RG&E Response:

The Ginna Station Reactor Protection System Logic includes a reactor trip on turbine trip (see drawing 882D612, sheet 2, sent to Dennis L. Ziemann on January 18, 1979).

Separation of HPCI and RCIC System Initiation Levels - Analysis and Implementation (II.K.3.13)

This item applies only to BWRs and is not applicable to R. E. Ginna.

Isolation of Isolation Condensers on High Radiation (II.K.3.14)

This item applies only to BWRs and is not applicable to R. E. Ginna.

Modify Break Detection Logic to Prevent Spurious Isolation of HPCI and RCIC Systems (II.K.3.15)

This item applies only to BWRs and is not applicable to R. E. Ginna.

Reduction of Challenges and Failures of Relief Valves - Feasibility Study and System Modification - (II.K.3.16)

This item applies only to BWRs and is not applicable to R. E. Ginna.

Report on Outage of ECC Systems - Licensee Report and Proposed Technical Specification Changes (II.K.3.17)

Position

Several components of the ECC systems are permitted by Technical Specifications to have substantial outage times (e.g., 72 hours for one diesel-generator; 14 days for the HPCI system). In addition, there are no cumulative outage time limitations for ECC systems. Licensees should submit a report detailing outage dates and lengths of outages for all ECC systems for the last five years of operation. The report should also include the causes of the outages (e.g., controller failure, spurious isolation).

Clarification

This item is applicable to all LWRs.

Discussion

NUREG-0626, Section F-3.5

Schedule

January 1, 1981

Action: (Class I)

(1) Licensee to provide results of evaluation to staff by scheduled date.

RG&E Response:

Administrative procedures A-52.4, Control of Limiting Conditions for Operating Equipment, and A-52.5, Control of Limiting Conditions for System Specifications, address the method of controlling the time period that components and/or systems are inoperable or out of specification. Outage times and the reasons for outages are documented in these procedures.

A report detailing the outage dates and length of outages for all ECC systems at Ginna will be submitted as required by January 1, 1981. The report will cover the past five years of operation and will address the causes of the outages.

Modification of ADS Logic - Feasibility Study and Modifications for Increased Diversity for Some Event Sequences (II.K.3.18)

This item applies only to BWRs and is not applicable to R. E. Ginna.

Interlock on Recirculation Pump Loops (II.K.3.19)

This item applies only to BWRs and is not applicable to R. E. Ginna.

Loss of Service Water for Big Rock Point (II.K.3.20)

This item is not applicable to R. E. Ginna.

Restart of Core Spray and LPCI Systems on Low Level - Design and Modification (II.K.3.21)

This item applies only to BWRs and is not applicable to R. E. Ginna.

Automatic Switchover of RCIC System Suction - Verify Procedures and Modify Design - (II.K.3.22)

This item applies only to BWRs and is not applicable to R. E. Ginna.

Confirm Adequacy of Space Cooling for HPCI and RCIC Systems - (II.K.3.24)

This item applies only to BWRs and is not applicable to R. E. Ginna.

Effect of Loss of AC Power on Pump Seals (II.K.3.25)

Position

The licensees should determine by analysis or experiment, on a plant-specific basis, the consequences of a loss of cooling water to the reactor recirculation pump seal coolers. The pump seals should be designed to withstand a complete loss of alternating current power for at least two hours. Adequacy of the seal design should be demonstrated.

Clarification

This item is applicable to PWRs.

Discussion

NUREG-0626, Section A-2.14

Schedule

January 1, 1982

Action: (Class I)

Licensee to provide results of evaluation to staff by scheduled date.

RG&E Response:

Although this topic is listed as being applicable to PWRs, the Westinghouse Owners Group has confirmed with the staff that this item applies only to BWRs and is not applicable to R. E. Ginna.

Provide Common Reference Level for Vessel Level Instrumentation (II.K.3.27)

This item applies only to BWRs and is not applicable to R. E. Ginna.

Study and Verify Qualification of Accumulators on ADS Valves (II.K.3.28)

This item applies only to BWRs and is not applicable to R. E. Ginna.

Study to Demonstrate Performance of Isolation Condensers with Non-Condensibles (II.K.3.29)

Position

If natural circulation plays an important role in depressurizing the system (e.g., in the use of isolation condensers), then the various modes of two-phase flow natural circulation, including non-condensibles, which may play a significant role in plant response following a small-break LOCA should be demonstrated.

Clarification

This item is applicable to all LWRs.

Discussion

NUREG-0626, Section F-4.5

Schedule

April 1, 1981

Action: (Class I)

(1) Licensee to provide results of evaluation to staff by schedule date.

RG&E Response:

Although this topic is listed as being applicable to PWRs, the Westinghouse Owners Group has confirmed with the staff that this item applies only to BWRs and is not applicable to R. E. Ginna.

Revised Small-Break LOCA Methods to Show Compliance with 10 CFR 50, Appendix K (II.K.3.30)

Position

The analysis methods used by NSSS vendors and/or fuel suppliers for small break LOCA analysis for compliance with Appendix K to 10 CFR Part 50 should be revised, documented, and submitted for NRC approval. The revisions should account for comparisons with experimental data, including data from the LOFT and Semiscale facilities.

Clarification

Clarifying information regarding the implementation of this action item will be provided in the forthcoming Draft 4 of NUREG-0660. This item is applicable to LWRs.

Discussion

NUREG-0565 (2.2.2.a)
NUREG-0611 (3.2.1.a)
NUREG-0626 (F-4.0)
NUREG-0635 (3.2.1.a, 3.2.5.a)

Schedule

January 1, 1982

Action: (Class I)

(1) Licensee to submit analysis model for staff approval by scheduled date.

RG&E Response:

The present Westinghouse small break evaluation model used to analyze loss of coolant accidents for R. E. Ginna is in conformance with Appendix K to 10 CFR Part 50. However, Westinghouse has indicated that they will, nevertheless, address the specific NRC items contained in NUREG 0611 in a model change scheduled for completion by January 1, 1982.

Plant Specific Calculations to Show Compliance with 10 CFR 50.46 (II.K.3.31)

Position

Plant-specific calculations using NRC-approved models for small break LOCAs as described in II. K.3.31 above, to show compliance with 10 CFR 50.46 should be submitted for NRC approval by all licensees.

Clarification

Clarifying information regarding the implementation of this action item will be provided in the forthcoming Draft 4 of NUREG-0660. This item is applicable to LWRs.

Discussion

NUREG-0565 (2.2.2.b)
NUREG-0611 (3.2.1.b)
NUREG-0626 (F-4.0)
NUREG-0635 (3.2.1.b)

Schedule

January 1, 1983 or one year after staff approval of LOCA analysis model.

ACTION: (Class I)

(1) Licensee to provide results of evaluation to staff in accordance with the schedule as indicated above.

RG&E Response:

RG&E will provide the results of an evaluation in a timely manner following staff approval of a LOCA analysis model.

Evaluation of Anticipated Transients with Single Failure to Verify No Fuel Failure (II.K.3.44)

Position

For anticipated transients combined with the worst single failure and assuming proper operator actions, licensees should demonstrate that the core remains covered or provide analysis to show that no significant fuel damage results from core uncovery. Transients which result in a stuck-open relief valve should be included in this category.

Clarification

This item is applicable to LWRs.

Discussion

NUREG-0626, Section F-4.3

Schedule

1/1/81

Action: (Class I)

(1) Licensee to provide results of evaluation to staff by scheduled date.

RG&E Response:

Although this topic is listed as being applicable to PWRs, the Westinghouse Owners Group has confirmed with the staff that this item applies only to BWRs and is not applicable to R. E. Ginna.

Evaluation of Depressurization with Other Than ADS (II.K.3.45)

This item applies only to BWRs and is not applicable to R. E. Ginna.

Response to List of Concerns from ACRS Consultant (II.K.3.46)

This item applies only to BWRs and is not applicable to R. E. Ginna.

Identify Water Sources Prior to Manual Activation of ADS (II.K.3.57)

This item applies only to BWRs and is not applicable to R. E. Ginna.

ATTACHMENT C

RG&E RESPONSE TO

ENCLOSURE 4

WORKER RADIATION PROTECTION IMPROVEMENTS

(Darrel Eisenhut letter dated May 7, 1980)

Control Room Habitility Requirements

Position

In accordance with action item III.D.3.4, Control Room Habitability, licensees shall assure that control room operators will be adequately protected against the effects of accidental release of toxic and radioactive gases and that the nuclear power plant can be safely operated or shut down under design basis accident conditions (Criterion 19, "Control Room," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50).

Clarification

All facilities that have not been reviewed for conformance with the following sections of the Standard Review Plan:

2.2.1-2.2.2 Identification of Potential Hazards in Site Vicinity;

2.2.3 Evaluation of Potential Accidents;

6.4 Habitability Systems;

shall perform the necessary evaluations and recommend appropriate modifications to meet control room habitability reuqirements. The following documents may be used for guidance in performing the required evaluations:

1. Regulatory Guide 1.78, "Assumptions of Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release."

2. Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release."

3. K. G. Murphy and K. M. Campe, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Design Criterion 19," 13th AEC Air Cleaning Conference, August, 1974.

The licensee's submittal shall include the results of the analyses of control room concentrations from postulated accidental release of toxic gases and control room operator radiation exposures from airborne radioactive material and direct radiation resulting from design basis accidents. The toxic gas accident analysis should be performed for all potential hazardous chemical releases occurring either on the site or within five miles of plant site boundary. Regulatory Guide 1.78 lists the chemicals most commonly encountered in the evaluation of control room habitiability but is not all-inclusive.

The DBA radiation source term should be for the LOCA containment leakage and ESF leakage contribution outside containment as described in Appendix A and B of Standard Review Plan Chapter 15.6.5. In addition, BWR facility evaluations should add any leakage from the main steam isolation valves (e.g., valve stem leakage, valve seat leakge, main steam isolation valve leakage control system release) to the containment leakage and ESF leakage following a LOCA. Other DBA's should be reviewed to determine whether they might constitute a more severe control room hazard than the LOCA.

In addition to the accident analysis results which should either identify the possible need for control room modifications or provide assurance that the habitability systems will operate under all postulated conditions to permit the control room operators to remain in the control room to take appropriate actions as required by General Design Criteria 19, the licensee should submit sufficient information needed for an independent evaluation of the adequacy of the habitability systems. Attachment 1 lists the information that should be provided along with the licensee's evaluation.

Action: (Class II)

Licensees should submit their responses to this request on or before January 1, 1981. Modifications needed for compliance with the control room habitability requirements specified in this letter should be identified and a schedule for completion of the modifications should be provided. Implementation of such modifications should be started without awaiting for the results of the staff's review. Additional needed modifications, if any, identified by the staff during its review will be specified to licensees by July 1981. All modifications must be scheduled for completion by January 1, 1983.

RG&E Response:

The Ginna facility's control room habitability design features are being currently reviewed by the NRC's SEP Branch under Topic II-1.C, Potential Hazards or Changes in Potential Hazards Due to Transportation, Institutional, Industrial and Military Facilities; and Topic VI-8, Control Room Habitability. NRC site visits were conducted in September, 1978 and March 1979, and information was submitted to the Commission on January 23, 1979 and December 6, 1979 relative to control room habitability. We are aware that independent analyses of toxic chemical and other off-site hazards have been initiated by the SEP reviewers. It is our understanding that a Staff assessment of Ginna SEP Topics II-1.C and VI-8 has been drafted and will be published upon completion of the most recent NRC reorganization. Therefore, no further action is planned on Ginna control room habitability pending the publication and review of this most recent staff assessment.

Task III.D.3 Worker Radiation Protection Improvement

A. Objective

Improve nuclear power plant worker radiation protection to allow workers to take effective action to control the course and consequences of an accident, as well as to keep exposures as low as reasonably achievable (ALARA) during normal operation and accidents, by improving radiation protection plans, health physics, inplant radiation monitoring, control room habitability, and radiation worker exposure data base.

B. NRC Actions

(omitted)

C. Licensee Actions

1. Radiation protection plans.

   a. Description: Operating reactor licensees will develop an RPP based on NRC guidance and propose a technical specification change. Following NRC review, the licensees will take corrective actions, as necessary, based on inspection findings.

   b. Implementation: Operating reactors will complete by 15 months after issuance of requirement by NRC; operating license applicants will complete before fuel loading or by 15 months after issuance of requirement by NRC, whichever is later.

   c. Resources: 1.0 my per reactor; $5,000 for printing RPP and related procedures.

RG&E Response:

We have established and conducted the Ginna Station radiation control program in accordance with the Ginna Radiation Control Manual, Administrative Procedure A-1. This document defines all major corporate radiation protection policies and commitments to regulatory requirements in areas of training, routine and emergency exposure control, radiation monitoring, record keeping, handling of radioactive materials and medical radiation emergencies. Implementing procedures have been written under the Ginna HP-, RD-, A-, ES- and SC-1 procedure series which detail the specific requirements of the Ginna radiation control program. We have also established formal respiratory protection and ALARA review programs and associated implementing procedures.

All NRC guidance and regulations pertaining to health physics improvements are continually reviewed and incorporated in the Ginna radiation control program as appropriate. NRC inspection results are also reviewed and corrective action taken as is necessary. This policy will be continued.

Therefore, pending further clarification on the requirement for a Radiation Protection Plan (RPP), we believe the intent of the RPP is met by our current radiation control program.

2. Health physics improvements: This is a Decision Group D item.

3. Inplant radiation monitoring.

   a. Description: Licensees must evaluate locations and ranges of radiodine monitors, provide results to NRC, and install new monitors as required. They must also comply with the rule on radiation monitoring instruments and the Regulatory Guide on air-sampling instruments.

   b. Implementation: Operating reactors and near-term operating license applicants must have radioiodine detection capability by January 1, 1980 or before fuel loading and must add area monitors and a low-background area for iodine analysis by June 1982; other operating license applicants and construction permit holders must comply by June 1982 or prior to licensing for operation, whichever is later.

   c. Resources: Evaluation of radioiodine detection capability will require 0.2 my per reactor, and the addition of monitors will require 0.2 my and $50,000 per monitor.

RG&E Response:

These inplant radiation monitoring requirements have been addressed in our response to the Short Term Lessons Learned.

4. Control room habitability.

   a. Description: Licensees must review control room habitability against specified guidance and make necessary modifications.

   b. Implementation: For operating reactors, reviews must be complete by January 1981, and modifications must be complete by January 1983; operating license applicants must schedule necessary modifications to achieve compliance before full-power operating construction permit holders must comply before an operating license is issued.

c. Resources: 2.0 my and $500,000 per reactor for operating reactors. Estimate one-tenth of this for NTOLs that are likely to be substantially in compliance with existing guidance.

RG&E Response:

See our response to Control Room Habitability requirements above.

5. Radiation worker exposure data base: This is a Decision Group D item.

D. Other Actions: None.

E. References

(omitted)