DISTRIBUTION
Central Files
ORAB Reading
GHolahan

MAY 24 1982

MEMORANDUM FOR: G. C. Lainas, Assistant Director
for Safety Assessment
Division of Licensing

THRU: T. A. Ippolito, Chief
Operating Reactors Assessment Branch
Division of Licensing

FROM: G. Holahan, Section Leader
Systems Section
Operating Reactors Assessment Branch
Division of Licensing

SUBJECT: GENERIC RECOMMENDATIONS BASED ON THE REVIEW OF THE JANUARY 25, 1982 STEAM GENERATOR TUBE RUPTURE EVENT AT GINNA

The May 3, 1982 memorandum from Harold Denton calling for the development of generic recommendations requested that members of the Ginna Task Force be involved to the extent practical. Since I was the team leader responsible for review of the Plant System Response, on the Ginna Task Force, I am taking this opportunity to present my recommendations relative to the generic implications of the Ginna event. These recommendations are presented in the enclosure. I have divided my recommendations into three categories to differentiate among those items which (1) support the need for continuing on-going programs, (2) support the need for modifications to on-going programs, or (3) support the need for new generic programs. As requested in the May 3, 1982 memorandum on this subject, I have also identified these recommendations as relating to: Plant Systems Response, Human Factors Considerations, Radiological Consequences, Organizational Response or Post-Event Activities.

G. Holahan, Section Leader
Systems Section
Operating Reactors Assessment Branch
Division of Licensing

Enclosure:
As stated

cc w/enclosure:
See next page

8206080064 820524
PDR ADOCK 05000244
P PDR

OFFICE	SL-ORAB:DL	C-ORAB:DL					
SURNAME	GHolahan:dn	TIppolito					
DATE	5/24/82	5/24/82					

cc w/enclosure:

D. Eisenhut
R. Mattson
G. Holahan
J. Lyons
O. Crutchfield
T. Speis
B. Sheron
W. Lanning
W. Minners

OFFICE							
SURNAME							
DATE							

ENCLOSURE

1. Recommendations Supporting the Need for Continuing On-Going Programs

1.1 Plant Systems Response

1.1.1 Re' 'or Vessel Level Measurement

Th. lack of a reactor vessel level measurement system significantly complicated the Ginna event. It was the presence of a steam bubble of unknown size in the reactor vessel upper head and the fear of increasing the size of that bubble that caused the reactor operators to delay termination of high pressure safety injection. It was the continuing safety injection which lead to the overfilling of the steam generator and the opening of the Steam Generator safety valve. Installation of a reliable reactor vessel level measurement system would significantly aid in managing SGTR events.

1.2 Human Factors Considerations

1.2.1 Review of SGTR with Concurrent Failure of Primary or Secondary Relief or Safety Valves

The Ginna event was an SGTR which included both primary and secondary system valve failures. The PORV failure to close was quickly and effectively dealt with but the leakage of the Steam Generator Safety Valve went unnoticed and the complications it introduced in handling the event were not appreciated by the plant operators. Problems of multiple failures, beyond the design basis assumptions, are being handled through the TMI Action Plan item I.C.1. This program requires operator training and emergency procedures for the more important and more likely of the possible multiple failure events. Completion of Action Plan item I.C.1 is an appropriate and sufficient generic response to this concern.

1.3 Radiological Consequences

No comments

1.4 Organizational Response

No comments

1.5 Post Event Activities

No comments

2. Recommendations Supporting Modifications to On-Going Programs

2.1 Plant Systems Response

2.1.1 Pressurized Thermal Shock

During the Ginna SGTR event the isolation of the steam generator with the ruptured tube plus the tripping of the reactor coolant pumps resulted in natural circulation flow in one loop and near stagnant conditions in the other loop. The injection of cold ECCS water into the stagnant loop resulted in a rapid decrease in the cold leg temperature. During the January 25, 1982 event the cold water in the RCS cold leg apparently never flowed into the reactor vessel creating a potential thermal shock problem. However, the Ginna event did identify an important phenomena which may not be receiving sufficient attention in the Pressurized Thermal Shock program, that is, the influence of steam generator isolation and flow stagnation on the potential for Reactor Vessel Thermal Shock. The action plan on USI A-47 should be modified to specifically address and resolve this issue.

2.1.2 Reactor Coolant Pump Trip Requirement

This has been a difficult on-going issue for several years. The Ginna SGTR event is only the latest event to be complicated by the requirement to trip the Reactor Coolant Pumps (RCP) on a suspected small break LOCA. I am fully aware that the suggestion to trip the RCP's is a Westinghouse-developed recommendation which the staff accepted. However, it appears clear that all of the complications of the January 25, 1982 event probably would have been avoided if the RCP's had not been tripped.

Since licensing credit for rapid manual action to trip the RCP's is not consistent with past licensing practice, it does not appear that manual RCP trip resolves the legal 10 CFR 50.46, Appendix K concerns; and in terms of safety significance, I believe that allowing continued RCP operations is desirable for the following reasons. First, the size, location and timing of small LOCA's which would exceed 2200°F is very limited. Therefore, such events would be expected to be quite rate. Second, the benefits of RCP operation in terms of heat removal capability, plant control and plant-transient understandability are great. Third, if a small LOCA did begin to lead to unacceptable consequences because of excessive inventory loss associated with RCP operation, the existing, approved procedures for responding to indications of inadequate core cooling would lead the plant operators to take the necessary corrective action (primary and secondary depressurization). Therefore, while long term resolution of this issue is continuing, the NRC interim position requiring RCP trip should be changed.

2.1.3 Steam Generator Overfill

The Ginna event resulted in an overfilling of the steam generator and a flooding of the main steam line up to the MSIV. Overfilling of PWR SG has also occurred in the past yet there continues to be considerable confusion relative to the requirement to analyse such occurrences. AEOD has raised this issue on several occasions and has stated that the main steam line is not designed for the loads associated with such flooding. A program is needed to review and document the full spectrum of concerns and requirements in this area and to determine the degree of compliance in operating plants. This could be done in the frame work of the recently begun USI in this area.

2.2 Human Factors Considerations

.2.1 Accident Monitoring

The instrumentation used to monitor the course of the January 25, 1982 event had several deficiencies including non-redundant monitoring of the RCS pressure, failure of the position recording for secondary relief and safety valves and no flow or valve position monitoring on RCS leakage path such as the letdown relief valve and the seal-return line relief valve. Implementation of Regulatory Guide 1.97 on operating reactors would resolve problems associated with monitoring important parameters such as RCS pressure, however, the guide may need to be modified to more fully address the monitoring of primary and secondary leakage during events. Therefore, Reg. Guide 1.97 should be reviewed relative to its effectiveness in this area.

2.2.2 Emergency Procedure Reviews

During the Ginna event the formation of a steam bubble in the reactor vessel upper head occurred but had not been expected by the plant operators. Upon analysis of the event it is clear that steam formation should have been expected. The problem appears to be associated with the reluctance of the NSSS vendors and the licensees to perform best-estimate, plant specific analysis on the development of emergency procedures. The review process for approving plant emergency procedures and guidelines should be modified to require plant specific analysis in the development or at least in the verification of emergency procedures.

2.2.3 Shift Technical Advisor (STA)

During the Ginna event the STA involved himself directly in the process of handling the event by reading the emergency procedure to the plant operators. This is clearly not the independent, thoughtful, "stand back" overview role originally intended. It appears from this and other events that the original STA concept is not being properly implemented. This subject needs to be carefully evaluated and actions need to be taken to clarify the intent and strengthen the requirement or to revise the present program.

2.3 Radiological Consequences

No comments

2.4 Organizational Response

No comments

2.5 Post Event Activities

No comments

3. Recommendations for New Generic Programs

The extensive reliance on "non-safety related equipment" during the Ginna event indicates that the FSAR and SRP reviews of safety related equipment has been done with much too narrow a view as to what equipment is considered. Regulatory Guide 1.97 will resolve many of the issues related to instrumentation but a program to identify truely safety related equipment based on operating experience and emergency procedure review is needed. This information should be reflected in SRP modifications where necessary. Decisions on operating plants should be made after the scope of the problem is better understood.

3.2 Human Factors Considerations

No comments

3.3 Radiological Consequences

3.4 Organizational Response

3.4.1 Extensive training is needed for an effective NRC incident response. Comments on this matter were developed and documented in a March 4, 1982 memorandum from W. Minners to H. R. Denton and R. C. DeYoung, "Improving the NRC Incident Response Technical Assessment Capability." My comments on this subject are included in that document.

3.5 Post Event Activities

3.5.1 The finding of loose parts and the associated damage in the Ginna Steam Generator and the prior history of undetected loose parts damage clearly indicate the need for periodic shellside, visual inspection of steam generators. Requiring such inspection at the presently planned outages for eddy-current testing appears to be a reasonable approach.

DISTRIBUTION
Central Files
ORAB Reading
GHolahan

MAY 24 1982

MEMORANDUM FOR: G. C. Lainas, Assistant Director
for Safety Assessment
Division of Licensing

THRU: T. A. Ippolito, Chief
Operating Reactors Assessment Branch
Division of Licensing

FROM: G. Holahan, Section Leader
Systems Section
Operating Reactors Assessment Branch
Division of Licensing

SUBJECT: GENERIC RECOMMENDATIONS BASED ON THE REVIEW OF THE JANUARY 25, 1982 STEAM GENERATOR TUBE RUPTURE EVENT AT GINNA

The May 3, 1982 memorandum from Harold Denton calling for the development of generic recommendations requested that members of the Ginna Task Force be involved to the extent practical. Since I was the team leader responsible for review of the Plant System Response, on the Ginna Task Force, I am taking this opportunity to present my recommendations relative to the generic implications of the Ginna event. These recommendations are presented in the enclosure. I have divided my recommendations into three categories to differentiate among those items which (1) support the need for continuing on-going programs, (2) support the need for modifications to on-going programs, or (3) support the need for new generic programs. As requested in the May 3, 1982 memorandum on this subject, I have also identified these recommendations as relating to: Plant Systems Response, Human Factors Considerations, Radiological Consequences, Organizational Response or Post-Event Activities.

G. Holahan, Section Leader
Systems Section
Operating Reactors Assessment Branch
Division of Licensing

Enclosure:
As stated

cc w/enclosure:
See next page

8206080064 820524
PDR ADOCK 05000244
PDR

OFFICE	SL-ORAB:DL	C-ORAB:DL					
SURNAME	GHolahan:dn	TIppolito					
DATE	5/24/82	5/24/82					

cc w/enclosure:

D. Eisenhut
R. Mattson
G. Holahan
J. Lyons
D. Crutchfield
T. Speis
B. Sheron
W. Lanning
W. Minners

OFFICE ▶							
SURNAME ▶							
DATE ▶							

1. Recommendations Supporting the Need for Continuing On-Going Programs

1.1 Plant Systems Response

1.1.1 Reactor Vessel Level Measurement

The lack of a reactor vessel level measurement system significantly complicated the Ginna event. It was the presence of a steam bubble of unknown size in the reactor vessel upper head and the fear of increasing the size of that bubble that caused the reactor operators to delay termination of high pressure safety injection. It was the continuing safety injection which lead to the overfilling of the steam generator and the opening of the Steam Generator safety valve. Installation of a reliable reactor vessel level measurement system would significantly aid in managing SGTR events.

1.2 Human Factors Considerations

1.2.1 Review of SGTR with Concurrent Failure of Primary or Secondary Relief or Safety Valves

The Ginna event was an SGTR which included both primary and secondary system valve failures. The PORV failure to close was quickly and effectively dealt with but the leakage of the Steam Generator Safety Valve went unnoticed and the complications it introduced in handling the event were not appreciated by the plant operators. Problems of multiple failures, beyond the design basis assumptions, are being handled through the TMI Action Plan item I.C.1. This program requires operator training and emergency procedures for the more important and more likely of the possible multiple failure events. Completion of Action Plan item I.C.1 is an appropriate and sufficient generic response to this concern.

1.3 Radiological Consequences

No comments

1.4 Organizational Response

No comments

1.5 Post Event Activities

No comments

2. Recommendations Supporting Modifications to On-Going Programs

2.1 Plant Systems Response

2.1.1 Pressurized Thermal Shock

During the Ginna SGTR event the isolation of the steam generator with the ruptured tube plus the tripping of the reactor coolant pumps resulted in natural circulation flow in one loop and near stagnant conditions in the other loop. The injection of cold ECCS water into the stagnant loop resulted in a rapid decrease in the cold leg temperature. During the January 25, 1982 event the cold water in the RCS cold leg apparently never flowed into the reactor vessel creating a potential thermal shock problem. However, the Ginna event did identify an important phenomena which may not be receiving sufficient attention in the Pressurized Thermal Shock program, that is, the influence of steam generator isolation and flow stagnation on the potential for Reactor Vessel Thermal Shock. The action plan on USI A-47 should be modified to specifically address and resolve this issue.

2.1.2 Reactor Coolant Pump Trip Requirement

This has been a difficult on-going issue for several years. The Ginna SGTR event is only the latest event to be complicated by the requirement to trip the Reactor Coolant Pumps (RCP) on a suspected small break LOCA. I am fully aware that the suggestion to trip the RCP's is a Westinghouse-developed recommendation which the staff accepted. However, it appears clear that all of the complications of the January 25, 1982 event probably would have been avoided if the RCP's had not been tripped.

Since licensing credit for rapid manual action to trip the RCP's is not consistent with past licensing practice, it does not appear that manual RCP trip resolves the legal 10 CFR 50.46, Appendix K concerns; and in terms of safety significance, I believe that allowing continued RCP operations is desirable for the following reasons. First, the size, location and timing of small LOCA's which would exceed 2200°F is very limited. Therefore, such events would be expected to be quite rate. Second, the benefits of RCP operation in terms of heat removal capability, plant control and plant-transient understandability are great. Third, if a small LOCA did begin to lead to unacceptable consequences because of excessive inventory loss associated with RCP operation, the existing, approved procedures for responding to indications of inadequate core cooling would lead the plant operators to take the necessary corrective action (primary and secondary depressurization). Therefore, while long term resolution of this issue is continuing, the NRC interim position requiring RCP trip should be changed.

2.1.3 Steam Generator Overfill

The Ginna event resulted in an overfilling of the steam generator and a flooding of the main steam line up to the MSIV. Overfilling of PWR SG has also occurred in the past yet there continues to be considerable confusion relative to the requirement to analyse such occurrences. AEOD has raised this issue on several occasions and has stated that the main steam line is not designed for the loads associated with such flooding. A program is needed to review and document the full spectrum of concerns and requirements in this area and to determine the degree of compliance in operating plants. This could be done in the frame work of the recently begun USI in this area.

2.2 Human Factors Considerations

.2.1 Accident Monitoring

The instrumentation used to monitor the course of the January 25, 1982 event had several deficiencies including non-redundant monitoring of the RCS pressure, failure of the position recording for secondary relief and safety valves and no flow or valve position monitoring on RCS leakage path such as the letdown relief valve and the seal-return line relief valve. Implementation of Regulatory Guide 1.97 on operating reactors would resolve problems associated with monitoring important parameters such as RCS pressure, however, the guide may need to be modified to more fully address the monitoring of primary and secondary leakage during events. Therefore, Reg. Guide 1.97 should be reviewed relative to its effectiveness in this area.

2.2.2 Emergency Procedure Reviews

During the Ginna event the formation of a steam bubble in the reactor vessel upper head occurred but had not been expected by the plant operators. Upon analysis of the event it is clear that steam formation should have been expected. The problem appears to be associated with the reluctance of the NSSS vendors and the licensees to perform best-estimate, plant specific analysis on the development of emergency procedures. The review process for approving plant emergency procedures and guidelines should be modified to require plant specific analysis in the development or at least in the verification of emergency procedures.

2.2.3 Shift Technical Advisor (STA)

During the Ginna event the STA involved himself directly in the process of handling the event by reading the emergency procedure to the plant operators. This is clearly not the independent, thoughtful, "stand back" overview role originally intended. It appears from this and other events that the original STA concept is not being properly implemented. This subject needs to be carefully evaluated and actions need to be taken to clarify the intent and strengthen the requirement or to revise the present program.

2.3 Radiological Consequences

No comments

2.4 Organizational Response

No comments

2.5 Post Event Activities

No comments

3. Recommendations for New Generic Programs

The extensive reliance on "non-safety related equipment" during the Ginna event indicates that the FSAR and SRP reviews of safety related equipment has been done with much too narrow a view as to what equipment is considered. Regulatory Guide 1.97 will resolve many of the issues related to instrumentation but a program to identify truely safety related equipment based on operating experience and emergency procedure review is needed. This information should be reflected in SRP modifications where necessary. Decisions on operating plants should be made after the scope of the problem is better understood.

3.2 Human Factors Considerations

No comments

3.3 Radiological Consequences

3.4 Organizational Response

3.4.1 Extensive training is needed for an effective NRC incident response. Comments on this matter were developed and documented in a March 4, 1982 memorandum from W. Minners to H. R. Denton and R. C. DeYoung, "Improving the NRC Incident Response Technical Assessment Capability." My comments on this subject are included in that document.

3.5 Post Event Activities

3.5.1 The finding of loose parts and the associated damage in the Ginna Steam Generator and the prior history of undetected loose parts damage clearly indicate the need for periodic shellside, visual inspection of steam generators. Requiring such inspection at the presently planned outages for eddy-current testing appears to be a reasonable approach.