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MINUTES OF THE ACRS SUBCOMMITTEE
MEETING ON REACTOR OPERATIONS/ R. E. GINNA
ROCHESTER, NY
MARCH 18-19, 1982

CERTIFIED COPY
ISSUE DATE: JULY 7, 1982

The ACRS Subcommittee on Reactor Operations and the R. E. Ginna Nuclear Power Plant met at the Sheraton Inn, 1100 Brooks Avenue, Rochester, NY, on March 18-19, 1982. The purpose of this meeting was to discuss the January 25, 1982 steam generator tube failure and site emergency incident, and the Systematic Evaluation Program (S.E.P.) as applied to Ginna. On March 18, 1982, the meeting was recessed at approximately 12:30 p.m. so that the Subcommittee could tour the plant. Notice of this meeting was published in the Federal Register on Monday, March 1, 1982. The Federal Register Notice is Attachment A. A copy of the detailed schedule of presentations is shown in Attachment B. The attendee list is Attachment C. The Subcommittee received written comments from Mr. Peter Mitchell, spokesman for the Rochester Safe Energy Alliance (comments are included as Attachment D). A complete set of presentation slides is on file in the ACRS offices. Attachment E is a list of slides presented. Mr. David Fischer was the Designated Federal Employee for this meeting. Portions of the meeting dealing with steam generator operating history; the results of the recent steam generator tube inspection, analysis, and testing; and the steam generator tube repair programs were closed to public attendees due to pending civil litigation related to these matters.

FIRST DAY - THURSDAY, MARCH 18, 1982

Introduction

Mr. W. Mathis, Subcommittee Chairman, opened the meeting with a brief statement on the purpose and goals of the meeting. He noted there were no requests to make oral statements, or written statements submitted. (Later in the meeting Peter Mitchell representing Rochester Safe Energy Alliance, requested permission to question Mr. Morris on the incident. It was suggested that questions in writing would be considered on the following day.) Dr. Robert Mecredy of Rochester Gas & Electric (RG&E) Corporation outlined the schedule for the two meeting days.

Plant Description

Mr. Bruce Snow (RG&E), Superintendent of the Ginna Station, provided the Subcommittee with a brief description of the plant site and plant systems. The Ginna Station is a Westinghouse 1520 megawatt-thermal pressurized water reactor. It drives a Westinghouse 496-MWe turbine generator. The plant is cooled by a direct cooling system from Lake Ontario. The pressurizer contains about 800 cu. ft. of volume. There are two sets of power-operated relief valves with motor-driven block valves in line on top of the pressurizer. The plant uses two Westinghouse series 44 steam generators. It is a two-loop plant. The emergency core cooling system is comprised of three intermediate pressure pumps and two low-pressure residual heat removal pumps.

Mr. Snow commented on Ginna's performance statistics. To date the Ginna Station has produced over 33 million MWe, the lifetime capacity factor has been 69%, and plant availability has been 75%. Key events in Ginna's history were described chronologically. The initial criticality was in the fall of 1969. Commercial operation began in July of 1970. In 1972 power was upgraded from 1300 MWt to 1520 MWt. The turbine building was modified to protect against floods, full flow condensate demineralizers were added, a new security building was built, and numerous TMI modifications have been added. Jet shields to protect against pipe breaks outside containment have been installed and a standby auxiliary feedwater system has been added in a separate structure containing two 200-gpm pumps. The plant in-service inspection program has been upgraded over the years.

Systematic Evaluation Program (SEP)-R. E. Ginna

Mr. George Wrobel, Senior Nuclear Engineer at Ginna, described the SEP as it applies to Ginna. He explained the SEP was a review of eleven nuclear plants, the oldest plants and those with provisional operating licenses. The purpose of the reviews was to review the plants against current regulatory requirements as expressed in the NRC Standard Review Plan. The SEP in Ginna's case would form part of the basis for a license conversion from a provisional operating license to a full-term operating license. The plan for the Ginna Station was begun in November of 1977 with 137 topics. Of the 137 topics, 45 were generic or not applicable to Ginna. Agreement between the NRC Staff and Licensee has been reached on approximately 75 of the remaining 92 topics (58 without modification, 1 with modification, and 16 with commitment to modify plant or procedures). Incomplete topics included several associated with low probability events, such as natural phenomena, or redundancy issues. Approximately $2 million has been spent on physical modifications as a result of the SEP, and $3 million for analyses and engineering. RG&E projects a total SEP cost in excess of $20 million (the original plant cost was $80 million). Two topics reviewed required expeditious correction of plant deficiencies and have resulted in new anchors for electrical equipment and a check valve leak test program prior to plant startups. Mr. Wrobel identified other physical modifications completed at Ginna; these include: battery room modifications to prevent flooding due to service water line cracks, seismically braced battery racks, and modified containment isolation logic. In addition, RG&E has initiated a piping seismic upgrade program independent of the SEP.

Mr. Wrobel listed the major analyses completed by RG&E and those completed by the NRC Staff or its consultants. RG&E SEP commitments to the NRC Staff were summarized and a detailed list of SEP open items was provided to the Subcommittee. Significant open items include:

- Wind and tornado loadings/combinations (RG&E study ongoing)
- Tornado and internally generated missiles (RG&E evaluating)
- Stability of slopes (RG&E analysis ongoing)
- Seismic analyses/modifications (RG&E analysis ongoing)
- Containment isolation valves (RG&E evaluating difference from GDC)
- Post-LOCA ESF switchover from injection to recirculation mode (RG&E to make modification, possibly only procedural)
- Station Service and Cooling Water Systems reliability (RG&E may need to make modifications in or to the screen house)

Ginna is currently in the initial phases of the Integrated Assessment portion of the SEP. The NRC Staff will be doing the Integrated Assessment for Ginna with input from RG&E.

RG&E's Appraisal of the SEP - R. Mecredy

Dr. R. Mecredy (RG&E) provided the Subcommittee with RG&E's appraisal of the SEP. He indicated that the SEP was beneficial for Ginna and cited several specific examples to support his statement. He stressed the importance of integrating plant modifications resulting from the SEP with modifications resulting from other reviews. The Ginna staff was questioned on the role of systems interactions in their SEP review. They noted a limited systems interaction study concerning the effect of failures of non-safety equipment on safety related systems.

NRC Staff Comments on Systematic Evaluation Program - W. Russell

Mr. W. Russell, Chief of the NRC's SEP Branch, provided the Subcommittee with a list of all SEP topics reviewed at Ginna which identified deviations from current licensing criteria. He gave a brief description of these differences. He introduced Mr. Allen Wang (NRR/SEPB) who identified 27 SEP topics which the NRC Staff currently considers open. Mr. Wang noted that the NRC's list of open items was longer than RG&E's list because the utility had eliminated those topics for which they had made proposals for resolution to the NRC Staff. Mr. Russell suggested several features that the Subcommittee should view on its site tour (e.g., service water pumps screen well house, auxiliary building, and condensate storage tank).

Mr. Russell explained the purpose of the integrated assessment process. He noted the aim was to review those differences that exist in the plant from current licensing criteria and to make basically two decisions. The first is to decide if a difference is significant enough to upgrade the facility, and, if it is, why? If the Staff's decision is that a topic is not significant enough to require an upgrade the Staff should document this second conclusion. If a particular topic does not require immediate or accelerated attention it is made part of the integrated assessment to be applied with other plant improvements to provide additional margins of safety as appropriate. The Staff doesn't intend to approve explicit design changes to the facility through the integrated assessment. Their goal is to identify areas to be upgraded, provide the utility an opportunity to produce an efficient design to address the concerns, and to provide a schedule for actual implementation.

Regarding the integrated assessment Mr. Russell noted the coordination between TMI Action Plan items and unresolved safety issues is not as close as once envisioned. He mentioned that the Integrated Assessment process was a joint process between the Staff and the utility. Mr. Russell then explained that as part of the Integrated Assessment an evaluation would be made of the risk reduction attendant with proposed plant modifications. Limited Probabilistic Risk Assessments (PRA) performed by Sandia will be used on both the Ginna and Palisades integrated plant assessments. However, Mr. Russell noted PRA is only one aspect of the assessment. The Staff is also considering safety significance on a deterministic judgmental basis. Both procedural and hardware modifications are being considered.

Response to the January 25, 1982 Steam Generator Tube Rupture at the Ginna Station - Art Morris

Mr. A. Morris, a senior reactor operator and member of the training department at the Ginna Station, described the steam generator tube rupture event. He described the personnel in the control room and the experience levels of each individual. He also mentioned that he was in the control room from ten minutes after the start of the incident until four o'clock in the afternoon (approximately 6½ hours). Mr. Morris felt that communications between the operations staff were very effective from both top-down and bottom-up. The control room manager was the shift supervisor who was in charge at all times. He was the pivot point and made communications within the control room effective, independent of the number of inputs.

Mr. Morris noted the procedures used during the incident were based on the Westinghouse Owner's Group Guidelines. He said that they were adequate. Today, based on the incident, there has been some fine-tuning of the procedures. Experience gained has been fed back to the Owners Group.

Mr. Morris suggested the response to the tube rupture can be described in three phases. The first phase is to diagnose the tube rupture. That was accomplished principally by the operators noticing the reactor coolant system pressure, and the pressurizer pressure and level were decreasing rapidly with no loss of coolant inventory into the containment building. The operators also saw an increasing radiation level on the air ejector and steam generator blow-down monitors. This was the major evidence that led operators to use the steam generator tube rupture procedure. Mr. Morris noted that it took twelve minutes to diagnose the problem and isolate the appropriate steam generator. The large primary to secondary flow rate which made the problem more obvious may have had some impact on how rapidly the operators responded.

The second phase in the response was leak stoppage. There were five basic steps which were procedurally guided. They were: identifying the faulted steam generator, isolating that steam generator; cooling down the reactor coolant system using the working steam generator; depressurizing the reactor coolant system to equalize the faulted steam generator (stopping the leak); and terminating safety injection pump operation to guard against repressurizing the reactor coolant system. Mr. Morris explained that it was during the depressurization and securing safety injection steps that the PORV stuck open. It was blocked within a minute of the time it stuck open when operators did not get a close indication on the control panel after signaling the PORV to close. The block valve was operated to stop the leakage out of the PORV. The depressurization of the plant took about thirty minutes; termination of safety injection was accomplished at the end of an hour and ten minutes.

The third phase of the incident response is the cooldown to cold shutdown. This was accomplished using the one good steam generator and starting the reactor coolant pump in the non-faulted loop. Eventually, temperature and pressure were reduced to the point where the residual heat removal system could be used. Steam generator tube repairs can begin when the plant is in a cold shutdown condition.

When questioned about the incident and the need or desire for any additional instrumentation, Mr. Morris expressed his personal opinion that a reactor vessel water level device would be useful, given its reliability. It was also noted that three core exit thermocouples that were previously withdrawn into the upper head region of the pressure vessel were used to detect the void that occurred in this region during depressurization.

Consequences of a Steam Generator Tube Rupture Event Compounded by a Stuck-Open Secondary Relief or Safety Valve - Mr. E. Volpenheim, Westinghouse

Mr. Eric Volpenheim (W) discussed the consequences of a steam generator (SG) tube rupture event compounded by a stuck-open valve on the secondary side (i.e., SG safety). The Westinghouse Owners Group is supporting an effort to analyze a design basis tube rupture coincident with stuck-open safety valves. Westinghouse has distributed emergency response guidelines for recovery from this event (to cold shutdown) for high-head safety injection (SI) plants. Guidelines for low-head SI plants are expected to be distributed in April-May of this year. The NRC Staff and INPO are reviewing these guidelines. Westinghouse is looking at the following issues as they relate to SG tube rupture events: reactor coolant pump trip and restart, SI termination, voiding of the reactor cooling system, long-term cooldown procedures, and SG overfill issues.

Radiological Emergency Organization - Mr. Lee Lang (RG&E)

Mr. Lee Lang (RG&E), Superintendent of Nuclear Production, provided the Subcommittee with a brief description of Ginna's offsite and onsite radiological emergency organizations. He highlighted the Ginna emergency organizational ties to the State of New York, Wayne County, and Monroe County. The information Ginna's emergency organizations provide to these government organizations was identified. These include information on the status of plant equipment and how the plant is operating, the status of any radioactivity released and meteorological information. Information goes via hot-line to the NRC, the State of New York, and local Counties (Wayne and Monroe). The procedures and facilities used by the Ginna emergency organizations were reviewed. Mr. Lang indicated that the NRC generally was not involved in the decision-making process during the event but noted that suggestions were solicited from the NRC during its onsite review committee meetings. In general, the plant's emergy system worked very well.

End of March 18, 1982 Subcommittee Meeting -

Following the recess, a tour of the plant was given to the Subcommittee by RG&E.

SECOND DAY - FRIDAY, MARCH 19, 1982

Introduction

Mr. W. Mathis, Subcommittee Chairman, opened the meeting and noted that Mr. Peter Mitchell, spokesman for the Rochester Safe Energy Alliance, had provided the Subcommittee with a list of questions regarding the presentations made by Mr. A. Morris (RG&E) the previous day. Mr. Mathis said that the Reactor Operations Subcommittee would consider these questions in its deliberations as the day's meeting progressed.

Schedule for March 19, 1982 - R. Mecredy, RG&E

Mr. R. Mecredy (RG&E) began the second day's proceedings with a brief description of the tentative schedule for the morning's meetings. He introduced Mr. Robert Witmer (Nixon, Hargrave, Devans & Doyle) counsel for RG&E. Mr. Witmer requested that portions of the meeting be closed due to pending civil litigation relating to the matters of steam generator operating history; the results of the recent SG tube inspection, analysis, and testing; and the SG tube repair programs.

Radiological Aspects of the Ginna Steam Generator Tube Rupture - Richard Watts (RG&E)

Mr. Richard Watts (RG&E) made a presentation on the radiological aspects of the Ginna steam generator tube rupture event. Mr. Watts explained that initially the lead responsibility for dose projection and radiological survey team direction comes from the technical support center. When proper communications flow and adequate staffing are achieved, the off-site Emergency

Operations Facility becomes the control point for conducting off-site radiological assessments. This arrangement allows the Plant Radiological Assessment group at the Technical Support Center to focus its attention more closely on in-plant concerns. Radiological and meteorological information was disseminated by direct telephone lines to the NRC, and by a direct hot line to the State of New York, and Wayne and Monroe Counties. The NRC provided helpful radiological support at both the TSC and off-site EOF.

Mr. Watts indicated that radionuclides were released from three points in the plant:

1. the condenser air ejector, which is combined with the gland-seal exhaust,
2. the turbine driven auxiliary feed pump exhaust and,
3. a relief valve on the steam line from the "B" steam generator.

Two types of releases occurred: noble gases, and radioiodines and particulates, including tritium. The total release of noble gases was estimated to be approximately between 30-42 Ci. The majority of noble gases were released through the air ejector. Estimates of radioiodine, particulates and tritium released from safety valve lifts are:

Total Iodine-131 Equivalent	0.16- 0.63 Ci
Total Particulates	0.3- 1.3 Ci
Tritium	5.9 - 24 Ci

Meteorological data was continuously collected on the Ginna site by instrumentation on two weather towers. A total of eight RG&E survey teams, each consisting of 2-3 persons, were used during the Ginna emergency.

Mr. Watts presented a summary of upper-bound offsite dose estimates from the Ginna tube rupture event. RG&E expects the more realistic dose values to be well under these upper-bound values. This conclusion is supported by environmental measurements to date. Even the upper-bound dose estimates fall within the current EPA standards for routine operations. It was explained that having timely and comprehensive environmental measurements ensured that correct measures were taken to protect the health and safety the public.

<u>Dissemination of Information to Industry and the General Public</u> - Lee Lang (RG&E)
Mr. Lee Lang (RG&E) identified some of the methods RG&E plans on using (or has already used) to disseminate to industry and the general public the information related to the Ginna SG tube rupture event. These include: the INPO notepad system, presentations to the Westinghouse Owners Group, and reports to the NRC. The NRC is also conducting an inquiry in which RG&E is cooperating. INPO and Westinghouse reviews are ongoing. Finally, presentations on the event will be made to the Edison Electric Institute and the American Nuclear Society.

<u>NRC Staff Position on Procedures Dealing with SBLOCA</u> - J. Lyons (NRC/NRR)
Mr. J. Lyons (NRC/NRR), Project Manager for the Ginna Nuclear Station, provided the Subcommittee with the NRC Staff's current position on RCP trip and HPCI termination subsequent to small break loss-of-coolant accidents.

He noted that based on analyses by the PWR NSSS vendors reactor coolant pumps are tripped in the event of a reactor trip and safety injection actuation, due to low system pressure. The Staff is evaluating the merits of automatic versus manual activation for RCP trip during a small break LOCA. The Staff does not recommend HPCI termination until a stable pressurizer level is established and the primary coolant attains 50° F. subcooling. (The 50° F. subcooling is not a rigorous requirement; it is intended to assure a degree of subcooling.)

Mr. Lyons also outlined the Staff's present review procedure related to limiting single failures during postulated events. This discussion specifically addressed the inability of plants to close a block valve subsequent to a SG atmospheric dump valve failing to reseat after lifting. He explained that double failures, such as the failure of an atmospheric dump valve plus the failure of its block valve to function, are not considered by the Staff.

Mr. R. Fraley, ACRS Executive Director, questioned the Licensee regarding plant cooldown rates during the SG tube rupture event. Cooldown rates during the event did exceed Technical Specification (T.S.) limits. The effects of this T.S. violation are being determined by Westinghouse. RG&E did explain that the Ginna highest pressure injection system has a maximum head of 1500 psi, which is in the intermediate pressure range. Pending completion of analyses by Westinghouse, RG&E expects that concerns over pressurized thermal shock occurring from repressurization of the system will be minimal.

NOTE: A transcript of the open portion of the meeting is on file at the NRC Public Document Room at 1717 H St., NW, Washington, DC or can be obtained from Alderson Reporters, 300 - 7th St., SW, Washington, DC, 292-554-2345.

kept, and questions may be asked only by members of the Subcommittee, its consultants, and Staff. Persons desiring to make oral statements should notify the Designated Federal Employee as far in advance as practicable so that appropriate arrangements can be made to allow the necessary time during the meeting for such statements.

The entire meeting will be open to public attendance except for those sessions which will be closed to protect proprietary information (Sunshine Act Exemption 4). One or more closed sessions may be necessary to discuss such information. To the extent practicable, these closed sessions will be held so as to minimize inconvenience to members of the public in attendance.

The agenda for subject meeting shall be as follows:

Tuesday, March 16, 1982

8:30 a.m. until the conclusion of business.

During the initial portion of the meeting, the Subcommittee, along with any of its consultants who may be present, will exchange preliminary views regarding matters to be considered during the balance of the meeting.

The Subcommittee will then hear presentations by and hold discussions with representatives of the NRC Staff, their consultants, and other interested persons regarding this review.

Further information regarding topics to be discussed, whether the meeting has been cancelled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by a prepaid telephone call to the cognizant Designated Federal Employee, Dr. Richard Savio (telephone 202/634-3267) between 8:15 a.m. and 5:00 p.m., EST.

I have determined, in accordance with Subsection 10(d) of the Federal Advisory Committee Act, that it may be necessary to close portions of this meeting to public attendance to protect proprietary information. The authority for such closure is Exemption (4) to the Sunshine Act, 5 U.S.C. 552b(c)(4).

Dated: February 24, 1982.

John C. Hoyle,

Advisory Committee Management Officer.

[FR Doc. 82-5416 Filed 2-26-82; 8:45 am]

BILLING CODE 7590-01-M

Advisory Committee on Reactor Safeguards; Subcommittee on Reactor Operations and R. E. Ginna; Meeting

The ACRS Subcommittee on Reactor Operations and R. E. Ginna will hold a meeting on March 18 and 19, 1982, at the SHERATON INN, 1100 Brooks Avenue, Rochester, NY. The Subcommittee will discuss the January 25, 1982 steam generator tube failure and site emergency incident and the Systematic Evaluation Program (SEP) as applied at Ginna. Notice of this meeting was published February 17.

In accordance with the procedures outlined in the Federal Register on September 30, 1981 (46 FR 47903), oral or written statements may be presented by members of the public, recordings will be permitted only during those portions of the meeting when a transcript is being kept, and questions may be asked only by members of the Subcommittee, its consultants, and Staff. Persons desiring to make oral statements should notify the Designated Federal Employee as far in advance as practicable so that appropriate arrangements can be made to allow the necessary time during the meeting for such statements.

The entire meeting will be open to public attendance except for those sessions during which the Subcommittee finds it necessary to discuss proprietary and Industrial Security information. One or more closed sessions may be necessary to discuss such information. (Sunshine Act Exemption 4). To the extent practicable, these closed sessions will be held so as to minimize inconvenience to members of the public in attendance.

The agenda for subject meeting shall be as follows:

Thursday, March 18, 1982—8:30 a.m. until the conclusion of business.

Friday, March 19, 1982—8:30 a.m. until the conclusion of business.

During the initial portion of the meeting, the Subcommittee, along with any of its consultants who maybe present, will exchange preliminary views regarding matters to be considered during the balance of the meeting.

The Subcommittee will then hear presentations by and hold discussions with representatives of the Rochester Gas and Electric Corporation, the NRC Staff, their consultants, and other interested persons regarding this review.

Further information regarding topics to be discussed, whether the meeting has been cancelled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by a prepaid telephone call to the cognizant Designated Federal Employee, Mr Richard K. Major (telephone 202/634-1414) between 8:15 a.m. and 5:00 p.m., E.S.T.

I have determined, in accordance with Subsection 10(d) of the Federal Advisory Committee Act, that it may be necessary to close some portions of this meeting to protect proprietary and Industrial Security information. The authority for such closure is Exemption (4) to the Sunshine Act, 5 U.S.C. 552b(c)(4).

Dated: February 24, 1982.

John C. Hoyle,

Advisory Committee Management Officer.

[FR Doc. 82-5417 Filed 2-26-82; 8:45 am]

BILLING CODE 7590-01-M

Advisory Committee on Reactor Safeguards; Subcommittee on Regulatory Activities; Change of Time

The March 3, 1982 meeting of the ACRS Subcommittee on Regulatory Activities has been rescheduled to start at 1:00 p.m. instead of 8:45 a.m. on March 3, 1982. All other items remain the same as published on Tuesday, Feb. 16, 1982, 31 FR 6741.

Dated: February 23, 1982.

John C. Hoyle,

Advisory Committee Management Officer.

[FR Doc. 82-5418 Filed 2-25-82; 8:45 am]

BILLING CODE 7590-01-M

Advisory Committee on Reactor Safeguards; Subcommittee on Waterford Steam Electric Station Unit No. 3; Change of Time

The March 3, 1982 meeting of the ACRS Subcommittee on Waterford Steam Electric Station Unit No. 3 has been rescheduled to start at 8:30 a.m. instead of 1:00 p.m. on March 3, 1982. All other items remain the same as published on Wednesday, February 17, 1982, 32 FR 7029.

Dated: February 23, 1982.

John C. Hoyle,

Advisory Committee Management Officer.

[FR Doc. 82-5419 Filed 2-26-82; 8:45 am]

BILLING CODE 7590-01-M

[Docket Nos. 50-400/401-OL]

Carolina Power & Light Co. and North Carolina Municipal Power Agency No. 3; Establishment of Atomic Safety and Licensing Board To Preside in Proceedng

Pursuant to delegation by the Commission dated December 29, 1972, published in the Federal Register (37 FR 28710) and §§ 2.105, 2.700 2.702, 2.711, 2.714a, 2.717 and 2.721 of the Commission's Regulations, all as amended, an Atomic Safety and Licensing Board is being established in the following proceeding to rule on petitions for leave to intervene and/or requests for hearing and to preside over the proceeding in the event that a hearing is ordered.

ATTACHMENT A

TENTATIVE SCHEDULE
ACRS SUBCOMMITTEE ON REACTOR OPERATIONS/GINNA
MARCH 18-19, 1982
ROCHESTER, NEW YORK

MARCH 18, 1982

8:30 A.M. I. Chairman's Introduction

A. Purpose of Meeting

B. Goals and Future Actions

8:45 A.M. II. Plant Description

9:15 A.M. III. Systematic Evaluation Program (SEP)

A. Current Status of the SEP at Ginna - outstanding items; current schedule for completion.

B. Any topics which required immediate resolution during the SEP review.

C. Which topics have or probably will require the greatest amount of analysis and/or modification to resolve?

D. What modifications have been made?

E. Preliminary views on cost vs. benefit judgments - how can they be made?

F. Licensee opinion of the SEP: worthwhile?; could it have been done in another way? How well integrated were SEP issues, Action Plan issues, and Unresolved Safety Issues?

10:30 A.M. * * * BREAK * * *

10:40 A.M. IV. Sequence of Events: January 25, 1982 Steam Generator Tube Rupture.

A. Event Chronology

1. Focused on major events

2. Rationale and methods for arriving at decisions as to course of action during the event.

ATTACHMENT B

B. Type of Emergency Operating Procedures Used - Event or Symptom Oriented?

C. Results of Emergency Planning Preparations. Ease with which initiated?

D. Coordination among licensee, NRC, FEMA, State and Local Agencies.

E. How useful were modifications to the plant as a result of the TMI experience.

1. Emergency Facilities

2. Shift Technical Advisor

3. Additional Plant Instrumentation

Additions that would have been useful not presently installed - Perspective of Operations Staff

12:30 P.M. RECESS Until 8:30 A.M. on March 19, 1982

- Lunch at Hotel

1:30 P.M. - Plant Tour

Leave hotel after lunch - 1:30 P.M. and drive to Ginna for Plant Tour.

MARCH 19, 1982

8:30 A.M. V. Steam Generator Tube Rupture: Jan. 25, 1982

A. Apparent Causes of the Steam Generator Tube Rupture: Proposed Fixes and Any Modifications.

B. Radiological Consequences of the Event.

C. Current NRC regulations on maintaining HPCI flow and RCP trip (NRC Staff)

D. Consequences of a steam generator tube rupture coupled with an inability to block the secondary steam dump valve. Any further consideration by Staff and Licensee.

E. Status of instrument air and RCP seal water flow on containment isolation.

F. Additional considerations given to steam voiding in the primary system.

F. Profile of operations staff: training and experience.

H. Industry Response and Involvement in Follow-Up Activities; INPO's role. How is the RG&E experience disseminated to the industry? Schedule for Additional Reports on the Event.

I. Lessons learned from the event

1. Any ways procedures could be improved?

2. Training improvements

3. Emergency planning improvements

4. Equipment modifications

12:30 P.M. Adjourn

MARCH 18-19, 1982
REACTOR OPERATIONS/R. E. GINNA SUBCOMMITTEE MEETING
ROCHESTER, NY
LIST OF ATTENDEES

ACRS

W. Mathis, Chairman
C. Siess, Co-Chairman
H. Etherington, Member
I. Catton, Consultant
D. Fitzsimmons, Consultant
R. Fraley, Executive Director, ACRS
D. Fischer, Designated Federal Employee

NRC STAFF

J. Lyons, NRR/DL/ORB-5
T. Martin, IE/RGN I
T. Su, NRR/DST/GIB
C. Petrone, IE/RGN I
W. Russell, NRR/SEPB
A. Wang, NRR/SEPB

OTHERS

J. Roberts, NYS-PSC
J. Keating, NYS-PSC
R. Leyse, EPRI, NSAC
P. Mitchell, RSEA
C. Marello
R. Consula, Recorder
R. Klak, Recorder
J. Dunkleberger, NYS Energy Office
J. Baranski, NYS REPG
T. Moore, Citizen
E. MacLaren, Citizen
C. Rosenthal, Concerned Citizen
M. Frank, UPI
G. Nobiling, Monroe Comm. College
M. Wert, Democrat & Chronicle
D. Persson, NYS Del.
A. Lowell, WSBY News

ROCHESTER GAS & ELECTRIC CO.

R. Mecredy
B. Snow
G. Wrobel
A. Morris
R. Peck
L. Lang
R. Watts
A. Curtis
L. Ermold
J. Hutton
R. Smith
J. Maier
J. Arthur
H. Rowley

NIXON, HARGRAVE, DEVANS & DOYLE

R. Witmer
P. Cronin

WESTINGHOUSE

T. Timmons

Attachment C

D.

3/19/82

FROM: Peter R. Mitchell - Spokesperson, Roch. Safe Energy Alliance
121 Edgerton St.
Rochester, N. Y. 14607
442-2929

TO: ACRS

TOPIC: Testimony by Art Morris on the Ginna Accident and Operator Response.

Questions are being submitted to the Committee through agreement and consideration given by the Hearing Chairman and Mr. Ray Fraley, ACRS Executive Director.

1. Background. The question was asked as to whether the existance of a water level indicator would have led to the operators responding differently with the HPIS and the PORV. Mr. Morris indicated this would be just one more piece of information and it would be second guessing. Hindsight is very important and can lead to both improved equipment and responses. The Themis P. Spies preliminary evaluation indicated, among other things, that two discharges of radioactive steam to the atmosphere from the faulted (B) generator occured as a result of HPI initially being left on longer than necessary and then being restarted at 11.15 a. m.

Questions. Why was the initial use of HPI not terminated when the reactor repressurized to Westinghouse guideline standards?

why was the HPI restarted at 11.15 a. m.?

In what manner did the 11.15 restart deviate from the Westinghouse guidelines?

Why wouldn't the existance of a water level indicator enable the operator to respond with greater precision in the use of HPI?

2. Background. The question was asked Mr. Morris as to whether the problem of reactor vessel thermal shock was considered during use of HPI. His answer was no. According to the Themis P. Spies preliminary report, the industry has indicated to the ACRS that operators would always terminate HPI before the primary system was unacceptably repressurized.

Questions. What repressurization perimeters did the Ginna operators use?

Has Westinghouse established guidelines regarding the thermal shock issue (both pressure and temperature)?

Did any of the reactor vessel cool at a rate in excess of that stipulated in the plant technical specifications.

If Westinghouse has not established guidelines regarding the thermal shock issue, are guidelines being contemplated, and,,if so, when will they be incorporated into the Ginna operating procedures?

ATTACHMENT D

3. Are there any conterplated changes in the design and operation of the PORV (due to the frequency with which they stick open)?

4. Were the emergency procedures in place at Ginna consistent with current Westinghouse Emergency Operator Guidelines for Steam Generator Tube rupture? If not, how were they different?

Are any changes contemplated by Westinghouse in their Guidelines and, if so, why?

Did the guidelines for response to a steam generator tube rupture contain instructions for actions to be taken xxxxxxxxxx when a steam bubble develops in the reactor vessel?

As a result of their experience with this accident, would the Ginna operators recom end any changes in the Westinghouse guidelines?

5. Is there any safety signifinance associated with stratification of the secondary coolant in the faulted (B) steam generator? Are any changes being recommended? If so, when will they be incorporated?

6. Background. The ACRS expressed a strong interest in learning more about reactor system interactions under accident conditions. The question was asked whether the Ginna operators had learned anything that would be helpful to others as far as procedures (both operator directions and technical based). Mr. Morris answered that little was learned from procedural directions on how to handle the accident, but technical based knowledge was gained and some changes have already been incorporated. He indicated this information has been provided the procedure subcommittee.

Question. What are the changes and/or suggested changes? How do I get a copy of this material?

7. Background. Mr. Morris indicated there were a number of instruments specific to the control room (besides a trustworthy water indicator guage) that would be helpful in dealing with future accidents. Since I was unable to accompany the ACRS on the tour, what instrumentation or modification of existing instruments would be helpful?

Questions for the ACRS.

In considering other coolant system failures and response scenarios, Theis P. Spies in his Preliminary Evaluation discusses two potential failures-- 1. tube leaks occuring in both steam generators simultaneously, and, 2. stuck open secondary side safety/relief valve.

In failure 1., the Westinghouse guideline recommends using the steam generator experiencing the smallest leak to cool the reactor. Is it possible to prevent releases of radiation to the atmosphere using this procedure? Is a feed and bleed a more desirable approach? Will the guidelines be changed to incorporate a feed and bleed approach?

Since a failure of the sss/rv can lead to core uncovering unless a) the valve is closed or b) additional cooling water supplies were made available, what steps are being taken to protect against this type of mode failure?

What caused the drop of the A generator pressure (less than 150 psi) with corresponding loss-of-condenser vacuum? What significance did this condition have in the accident sequence? Is any remedial action recommended?

ACRS Reactor Operations/R. E. Ginna Subcommittee Meeting
March 18-19, 1982
Rochester, N. Y.

List of Slides

Day 1 - 3/18/82

1. Ginna Nuclear Power Plant, ACRS Subcommittee Meeting March 18, 1982-Schedule
2. Schematic of RG&Es Ginna Nuclear Power Plant
3. Site Layout, Ginna Nuclear Power Plant
4. RGE, History Ginna Station (2 slides)
5. Systematic Evaluation Program - R.E. Ginna (8 slides)
6. Letter D. Crutchfield, NRC, to J. Maier, RG&E, Subject: Integrated Assessment Meeting at NRC Bethesda, dated March 17, 1982. Concerns reviews of S.E.P topics.
7. Slides used by A. Morris, RG&E, Steam Generator Tube Rupture Incident (12 slides)

Day 2 - 3/19/82

8. Slides used by R. Watts, RG&E: Radiological Assessment of the 1/25/82 Steam Generator Tube Rupture Event. 12 slides

9. Slides used by L. Lang, RG&E: Dissemination of Information to Industry and General Public

10. Slides used in closed session are all available with the exception of one slide which contains Confidential material, Ginna Station, B-Steam Generator, ACRS Mtg. (31 public slides, 1 Confidential slide)

Attacnment E