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Minutes of the
June 30, 1982 ACRS SEP Subcommittee Meeting
Integrated Plant Safety Assessment of the R. E. Ginna
Nuclear Power Plant

Washington, D. C.

The ACRS SEP Subcommittee met at 1717 H Street N. W., Washington, DC in Room 1046 on June 30, 1982. The purpose of this meeting was to review the activities of the Staff and their safety evaluation report related to the integrated plant safety assessment as a part of the Systematic Evaluation Program review of the Ginna plant. Notice of this meeting was published in the Federal Register on Wednesday, June 9, 1982. The Federal Register Notice is Attachment A. A copy of the schedule of presentations is shown in Attachment B. The attendee list is Attachment C. A complete set of presentation slides and meeting handout material is on file in the ACRS office. Attachment D is a list of meeting slides and handouts. Richard Major was the designated federal employee for this meeting. The entire meeting was open to the public.

Introduction:

Dr. Siess began the meeting by noting that the purpose of the meeting was to review the activities of the NRC Staff and their Safety Evaluation Report related to the integrated plant safety assessment review of the Ginna plant. He explained this was the second SEP review; Palisades was the first plant reviewed. He also explained that Ginna was included in the SEP, at least in part, so that the SEP review can form part of the basis for conversion from a provisional operating license to a full term operating license.

DESIGNATED ORIGINAL

Certified By BJR

8208290006

The ACRS will review the Ginna license conversion application at a later date. The license conversion will take into account methods for dealing with TMI Action Plan items, Unresolved Safety Issues, and the Systematic Evaluation Program.

Introduction By NRC Staff - C. Grimes

Mr. Grimes explained that the Staff is currently developing an implementation program for POL to FTOL conversions. This program will address the status of implementation of TMI action plan items and the status of Unresolved Safety Issues or the basis for continued operation relative to these issues. The status of multi-plant actions that have been identified as applicable to the plant under question, conformance with new regulations instituted since the issuance of the POL, and contentions raised if a hearing is involved in a Licensee conversion will be addressed. Resolution of outstanding items resulting from the SEP will be given. When plans for this program are completed, the Staff will describe their proposed course of action to the Committee.

Mr. Grimes stated that a schedule for POL to FTOL conversions has not been set. Dr. Siess suggested that the ACRS license conversion process revert back to the individual plant subcommittees rather than the SEP Subcommittee. He felt that the advantage of having one subcommittee review the license conversion will diminish, due to the more plant specific nature of the license conversions. The first license conversion will not occur until some time in 1983.

Mr. Grimes also explained that reviews for the next three SEP plants (Oyster Creek, Dresden 2, and Millstone 1) are being conducted in parallel.

All three plants use a Boiling Water Reactor. The Staff will issue integrated assessments about one month apart. Dr. Siess suggested that it may be possible to combine the ACRS review of two or all three of these plants, depending on the similarity of the reviews. All three plants have provisional operating licenses.

In commenting on the degree of integration in the SEP, Mr. Russell, Chief of the SEP Branch, noted that the Staff considered in the integrated assessment as much of the status of TMI Action Plan items and Unresolved Safety Issues as possible. In future Safety Assessment Reports the Staff will try to identify, to the extent practical, how other items and issues relate to SEP topics. In some cases, Licensees will evaluate criteria for upgrades, perform an evaluation of the safety benefits and costs associated with the upgrade, and return to the Staff for approval of a course of action.

The Staff noted that where immediate fixes are required Licensees have readily volunteered to make those fixes. (Ginna required no immediate fixes.) Integrated assessment items proposed to the Commission for implementation could, if necessary, be mandated by order. The Staff makes no distinction between a plant with a provisional operating license and a plant with a full term operating license regarding the regulatory basis for implementing fixes.

It was noted that some regulations (such as GDC 56 concerning containment isolation valves) allow the requirements to be met on a basis other than

that defined in the regulation. Where there is no flexibility in the letter of the regulations, and the current design satisfies the Staff, items will have to be identified as exceptions to the regulations.

Introduction by Licensee - Rochester Gas and Electric: Operating History and Plant Description for the R. E. Ginna Nuclear Power Plant - Mr. J. Larizza

The initial criticality at Ginna was in November of 1969. The initial power level was 1300 MWt. The plant began commercial operation in July 1970. In 1972, the power level was increased to 1520 MWt and RG&E applied for a full term operating license. In 1975, standby auxiliary feedwater systems were added and an upgrade to inservice inspection was made. Full flow condensate demineralizers were added and the SEP was started in 1977. Plant security upgrades were added in 1978. In 1980, TMI modifications, including a Technical Support Center, were added to the plant.

Ginna's performance statistics for plant life to date include: 33,853,098 MWe generated, a capacity factor of 69% and a plant availability of 75%. The Ginna reactor is a two loop Westinghouse PWR. It produces 1520 MWt (490 MWe). A unique feature of the Ginna plant is the total of five auxiliary feedwater pumps. Two are motor driven and can be operated off the diesels, one is a turbine driven pump. In addition, there are two standby auxiliary feedwater pumps that can inject water into the steam generators if normal auxiliary feedwater is not available. The standby AFW pumps take suction from the service water. They can be manually connected to the emergency diesel power.

Discussion of SEP Topics -NRC Staff - Alan Wang

Mr. Wang began his discussion with an overview of the status of the SEP topics. Of the 137 SEP topics, 24 were deleted since they were being reviewed generically under either the TMI Action Plan or as a USI; 21 additional topics were deleted that did not apply to the Ginna plant. Of the 92 remaining topics, 58 topics either met current criteria or were acceptable on another defined basis. Seven more topics were found acceptable after modifications made during the review. Twenty seven topics were considered for backfit in the integrated assessment.

Discussion of SEP Topics Found Acceptable on Another Defined Basis:

The Subcommittee reviewed with the Staff several examples of SEP topics that did not meet the exact letter of a particular requirement, but were found acceptable on another defined basis.

Dr. Thomas Cheng discussed the resolution of SEP Topic II-4, Geology and Seismology. Ginna's original seismic design was based on a Housner ground response spectrum with a peak ground acceleration of 0.2g. When the SEP seismic review of Ginna was begun, the site-specific ground response spectrum was not available to the Staff. The Staff chose the Regulatory Guide 1.60 spectrum scaled to the original 0.2g peak ground acceleration to begin their seismic review. The postulated SSE derived from this response spectrum is more conservative than either the site-specific spectrum or the original Housner spectrum. Following the Staff's review, two areas were found to be overstressed: some of the bracing in the auxiliary and turbine buildings. Questions related to the Unresolved Safety Issue on seismic

qualifications dealing with operability of equipment during an event will be covered in owners group responses to the USI; structural and mechanical integrity were covered under the SEP; functional integrity will be covered under a non-SEP generic item.

The Staff explained that SEP topic V-6 on reactor vessel integrity relates to the USI on pressurized thermal shock. Pressurized thermal shock will be handled by the Staff outside the scope of the SEP. Under the SEP topic, the Staff did review vessel material properties under current regulations in 10 CFR 50, Appendices G and H.

The Subcommittee examined the SEP review of topic III-4.B Turbine Missiles. The Staff based their review on the reliability of the overspeed turbine trip, and the adequacy of the inservice inspection of turbine discs. The Staff concluded that, for an interim period until a decision is reached regarding the need for updated probabilistic analysis of the turbine missile hazard, the probability of damage from turbine missiles is acceptably low.

The Subcommittee also discussed SEP topic XV-19, "Loss-of-coolant accidents resulting from spectrum of postulated piping breaks within the Reactor Coolant Pressure Boundary." The concern was centered on requiring the addition of sodium hydroxide (NaOH) almost immediately on the initiation of containment spray water. The Licensee noted that in the event of an inadvertent containment spray actuation, the difference between the clean-up problems presented by the otherwise acidic borated spray water or the alkaline spray water containing NaOH was negligible. (It would be a cleanup problem in either case.)

As a final example of topics resolved on other defined bases, the Subcommittee examined SEP Topics V-11-A and V-11.B, "Requirements for Isolation of High and Low Pressure Systems." The safety objective of the topic is to assure the integrity of high pressure systems such as the reactor coolant system and to maintain the availability of low pressure systems such as the residual heat removal (RHR) and chemistry and volume control (CVCS) systems. The conclusions reached by the Staff were that RHR valves do not have to close automatically on increasing pressure, since this isolation would prevent the proper functioning of the overpressurization protection system designed to mitigate this event. It was also determined, based on a valve testing program, that no single failure can cause overpressurization of the RHR system; therefore, additional RHR interlocks are not required. The Staff also reached the conclusion concerning the CVCS that a failure of the CVCS letdown non-regenerative heat exchanger inlet valve may not be automatically isolated. However, radiological consequences are acceptably low provided an operator responds in a timely manner to area radiation monitor alarms by closing the CVCS orfice isolation valve.

Senior Review Group

During the meeting, the SEP Branch passed out copies of comments on the Ginna integrated plant safety assessment by the NRC Staff's Senior Review Group. Members of the Senior Review Groups include: J. Hendrie, S. Bush, Z. Zudans, H. Isbin, and R. Budnitz. In general, the comments were favorable on the methods used in the Ginna integrated assessment.

Integrated Assessment of 27 Topics Considered for Backfit A. Wang, NRC Staff

The topics considered for backfit were divided into five categories for presentation to the Subcommittee. The categories included topics not requiring backfit, topics with procedural backfits, topics with hardware backfits, topics with analysis and potential hardware backfits, and topics with differences between RG&E and the Staff.

The items the Staff concluded did not require backfit were shown to the Subcommittee. These items included:

- Topic II-4.D, Stability of Slopes

- Topic III-4.A, Tornado Missiles
  (Section 4.11.3, Boric Acid Tanks)

- Topic III.4.C, Internally Generated Missiles
  (Sections 4,12.1, Accumulator (CVCS) Letdown Lines and 4.12.4, Refueling Water Storage Tank)

- Topic III-6, Seismic Design Consideration
- (Section 4.15.2, Turbine Bldg.)

- Topic III-8.A, Loose Parts Monitoring and Core Barrel Vibration Program

- Topic V-5, Reactor Coolant Pressure Boundary Leakage to Containment
  (Section 4.15.1, Detection of Reactor Coolant Pressure Boundary Leakage to containment)
  (Section 4.19.2, Monitoring of Reactor Coolant Inleakage)
  (Section 4.19.3, Technical Specifications Regarding Operability of Leakage Detection Systems)
  (Section 4.15.4, Reactor Coolant Inventory Balance)

- Topic VI-4, Containment Isolation System
  (Section 4.22.1, Valve Location)
  (Section 4.22.2, Valve Number)
  (Section 4.22.3, Valve Actuation)

- Topic IX-3, Station Service and Cooling Water Systems
  (Section 4.25,4, Pressure Sensor on Component Cooling Water Pumps)

- Topic IX-5, Ventilation System

The Subcommittee next took up the topics which are the subject of additional analysis and potential hardware backfit. These topics included:

- Topic III-1, Classification of Structures, Systems and Components (Seismic and Quality Group) - This issue generally concerns the amount of nondestructive testing to be performed such as radiography of certain welds.

- Topic III-4.A, Tornado Missiles (Section 4.11.1, Component Cooling Water System)

Here the concern is reaching cold shutdown without using the component cooling water system which is assumed lost from tornado missile damage. The Licensee will make analyses to show it is possible to reach cold shutdown without the CCW. An exception will probably be required since it will take longer than the 10 CFR 50 Appendix R requirement that cold shutdown be achieved in 72 hours.

- Topic III-5.A, Effects of Pipe Break on Structures, Systems and Components Inside Containment

The Staff noted that this topic remains open as break locations are still being postulated. Jet impingment, pipewhip, flooding effects, etc., are being analyzed for safety-related components.

- Topic III-6, Seismic Design Consideration (Section 4.15.4, Safety Related Tanks, Section 4.15.5, Electrical Panels Section 4.26.6, Ability of safety-related electrical equipment to function, Section 4.15.7, Qualification of cable trays)

In general, the Licensee's seismic analysis is still underway. In some cases, owner group reports are expected as well as some equipment testing.

- Topic III-7.B , Design Codes, Design Criteria, Load Combinations, and Reactor Cavity Design Criteria.

The Licensee under this topic is performing additional analysis on structural loadings on the containment liner insulation.

- Topic IX-3, Station Service and Cooling Water Systems (Section 4.25.3, Flooding Due to Failure of Tanks)

For this item the Licensee proposes to evaluate the design of various tanks in the auxiliary building.

- Topic IX-6, Fire Protection

This topic and associated exemption requests are being resolved as part of the Appendix R reviews. RG&E is proposing to install a dedicated shutdown system that is completely independent of the control room to aid in meeting the requirements of Appendix R to 10 CFR 50.

The Subcommittee next considered items that will require procedural or administrative changes. The following topics required procedural backfits. Included is a brief explanation of the nature of the backfit.

- Topic II-1.A. Exclusion Area Authority and Control - This topic will require that the current exclusion area boundAry map be incorporated in the Ginna tech. specs.

- Topic II.3.C, Inservice Inspection of Water Control Structures
  The Licensee will modify their inspection program as recommended by the staff with the exception of Deer Creek which remains an open issue.
  (Equipment such as roof drains are included as water control structures.)

- Topic III.7.A, Inservice Inspection Including Prestressed Concrete Containments with either grouted or ungrouted tendons - the Licensee will modify their surveillance program as recommended by the Staff.

- Topic V-10.B, Residual Heat Removal System Reliability - This item will require the development of procedures for operation of safety-grade systems and components to achieve cold shutdown if non-safety-grade systems are unavailable. This item also involves procedures for overpressure protection of the shutdown cooling systems. Whenever the RHR system is running, the overpressurization protection system will be activated.

- Topic VI-7.B, Engineered Safety Feature Switchover from Injection to Recirculation Mode. The Licensee has contracted Westinghouse to review procedures and improve the switchover procedure.

- Topic IX-3, Station Service and Cooling Water Systems - Currently the plant can be operated with the minimum number of SWS pumps aligned to one bus. Technical specifications will be modified to correct this by adding diversity to pump alignment.

The Subcommittee next discussed items that will require hardware backfits.

The following items list the topics of concern along with the backfit.

- Topic III.4.C, Internally Generated Missiles - This item will require a restraint for a valve operator.

- Topic 11-2.A, Severe Weather Phenomena, III-2, Wind and Tornado Loadings, III-4.A, Tornado Missiles, III-6, Seismic Design Consideration, II-7.B, Design Codes, Design Criteria, Load Combinations, and Reactor Cavity Design Criteria - Structures important to safety do not meet all current licensing criteria. The Licensee will develop design parameters and criteria for structural upgrades, perform the structural analysis and engineering design of proposed modifications, and install modifications, as required, as a result of the analysis.

- Topic III.5.B, Pipe Break Outside Containment and Topic III-6, Seismic Design Consideration - Service Water Pumps are susceptable to several common mode failures (seismic, fire, wind loading, pipe break and flooding). The Licensee has agreed to provide a backup cooling water source for the diesel generator and upgrade the essential service water system.

- Topic V-10.A, Residual Heat Removal System Heat Exchange Tube Failure-Licensee will install a radiation monitor for the service water system or include surveillance and operability requirement for the CCW System radiation monitor.

- Topic VIII-3.B, DC Power System Bus Voltage monitoring and annunciation The Licensee has agreed to provide aditional DC System monitoring and a "DC System Trouble Alarm."

- Topic IX-3, Station Service and Cooling Water System - A second transmitter and level alarms will be installed on the component cooling water surge tank to ensure adequate water level indication.

The Subcommittee discussed topics with differences between RG&E and the Staff. The course to resolution of these items is still not set. In some cases additional analyses will be required. In all three cases below the Licensee has not submitted a written position on the issue to the staff. The three areas of controversy include: flooding potential, the effects of groundwater level on structures, and the containment isolation system.

The issue on flooding concerns the flood level associated with Deer Creek, a small stream flowing into Lake Ontario in the vicinity of the plant. Flooding from Deer Creek was not considered when the plant was originally licensed. Current design practice calls for evaluating flood levels based on the Probable Maximum Flood. This flood would result in water levels four feet deep at grade at high portions of the plant. At lower portions of the plant next to the rear wall, screen house, service water pumps, diesel generators, and vital buses, the water level could be 12 feet above grade. The staff has been unwilling to backfit the PMF to the Ginna site. Instead, the staffhas taken the position that the plant must provide protection for the Standard Project Flood plus 1 ft. This flood amounts to a flood three times greater than the Historically land flood, but about one third from the PMF. The Licensee and the Staff are still at odds over the capacity of Deer Creek to drain without flooding given the Standard Project Flood.

The Subcommittee covered next the subject of high groundwater level on structures. The area of disagreement relates to the groundwater level and the associated hydrostatic pressures that structures below grade must withstand. The plant was designed assuming a groundwater elevation of 250 ft. Although limited observations from borings have shown the groundwater

withstand. The plant was designed assuming a groundwater elevation of 250 ft. Although limited observations from borings have shown the groundwater to be near that elevation, there has been no program of continuing measurements to demonstrate that the level does not exceed 250 ft. during periods of prolonged precipitation. Without such a program, the NRC Staff has determined that the effects of groundwater should be evaluated at the grade elevation, approximately 270 ft., for the structures of interest. The Staff has determined that the effects of groundwater at grade on all safety related structures need to be evaluated. Groundwater loading becomes significant when it is part of a load combination with seismic loadings. The Licensee contends that 13 years of operation has shown the present design adequate.

Another area of disagreement between the Staff and Licensee is the subject of certain containment isolation valves. These valves do not satisfy the requirements of GDC 57. The containment penetrations in question involve lines in closed systems that are neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere. These lines presently contain manually operated valves whereas GDC 57 would require the valves to be remotely operated and able to pass a leak rate test as isolation valves. Although the NRC Staff has accepted "other defined bases" for not requiring strict compliance with the GDC for many other isolation valves in the Ginna Plant, they have listed several reasons for requiring conformance in these particular cases. Some of the Staff's reasons are: the service water lines in question are large, system pressure at the outlet is lower than accident pressure, and access to present manual valves may be limited due to high radiation.

The Licensee objects to the isolation requirements on technical grounds as well as the substantial cost to convert (replace) the valves to remote-manual operation. The Licensee feels that dose rates at the valves following an accident will be low enough to allow manual activation. The Licensee also feels there is sufficient time to isolate the system in question (service water to and from fan coolers) and avert any appreciable release of radiation given a leak. The additional information supplied by the Licensee during this meeting will be considered by the Staff.

Limited Probabilistic Risk Assessment

The limited PRA performed for Ginna was briefly discussed. The PRA was based on the results of a combination of WASH-1400 and the Crystal River IREP study. In particular, the risk reduction potential of the leakage detection system was discussed. The importance of this system was rated high, however, this rating was based on incorrect assummptions made by those doing the assessment. The sensitivity of the system was incorrectly assumed to be one gallon in 12 hours, when in fact, the sensitivity is one gallon in one hour. This difference in sensitivity affects the amount of time necessary to detect and then stop a pipe leak before it becomes a break. If this information had been in hand at the time of the analysis, this issue would not have received a rating of high importance. Local and global plant leakage detection is still under study to see if the composite of nine leak detection systems is adequate without augmentation.

Discussion by Licensee on the Worth of the SEP/Integrated Assessment as Applied to Ginna -R. Mecredy, RG&E

Dr. Mecredy of Rochester Gas & Electric addressed the worth of the SEP. He felt the objective to reassess the safety margins has largely been met. The objective to create a documentation base has not been completely met. The objective to provide the compatability for an integrated assessment has not been met. It does provide the basis for future integration. The objective of this utility participating in SEP is to convert the provisional operating license to a full term operating license. That objective continues to be elusive.

Dr. Mecredy felt that for phase III of the SEP the number of important topics should be about 10 to 20 for detailed reviews. The total amount of time necessary for the entire review was felt to be about two years.

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, D. C. or can be obtained from Alderson Reporters, 300 7th St. S.W., Washington, D. C. 202-554-2345

Technical Information and Document Control, Washington, D.C. 20555, Attention: Publications Unit.

Dated at Bethesda, Maryland, this 27th day of May 1982.

For the Nuclear Regulatory Commission.

Dennis M. Crutchfield,

Chief, Operating Reactors Branch No. 5, Division of Licensing.

[FR Doc. 82-15607 Filed 6-8-82; 8:45 am]

BILLING CODE 7590-01-M

Ad Hoc Committee for Review of Nuclear Reactor Licensing Reform Proposals; Changed Meeting

This is to announce a change in starting time for the next meeting of NRC's Ad Hoc Committee for Review of Nuclear Reactor Licensing Reform Proposals to be held on June 10, 1982. The starting time will be 9:30 a.m. instead of 10:00 a.m. as previously announced in the Federal Register (47 FR 21168). All other information regarding the June 10 meeting remains unchanged.

Subsequent meetings of this Ad Hoc Committee are now scheduled for June 28 and July 12, 1982. The meetings will be open for public observation and will begin at 9:30 a.m. on both days at the offices of Shaw, Pittman, Potts and Trowbridge, South Building, 9th Floor Lobby, 1800 M St., N.W., Washington, D.C.

At these meetings, the Committee will continue its review of proposals for reforming the NRC's licensing process for nuclear plants. Transcripts of the meetings will be made available for public inspection and copying at NRC's Public Document Room, 1717 H St., N.W., Washington, D.C.

Further information on the meetings may be obtained from Mr. Rothschild, Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555 (Tel. 202/634-1465).

Dated at Washington, D.C., this 3rd day of June 1982.

John C. Hoyle,

Advisory Committee Management Officer.

[FR Doc. 82-15610 Filed 6-8-82; 8:45 am]

BILLING CODE 7590-01-M

Advisory Committee on Reactor Safeguards, Subcommittee on Systematic Evaluation Program; Meeting

The ACRS Subcommittee on the Systematic Evaluation Program will hold a meeting on June 30, 1982 in Room 1046, 1717 H Street, NW., Washington, DC. The Subcommittee will review the Integrated Plant Safety Assessment, Systematic Evaluation Program review of the R. E. Ginna Nuclear Power Plant.

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 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: *Wednesday, June 30, 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 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 Major (telephone 202/634-1414) between 8:15 a.m. and 5:00 p.m. e.d.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 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: June 4, 1982.

John C. Hoyle,

Advisory Committee Management Officer.

[FR Doc. 82-15611 Filed 6-8-82; 8:45 am]

BILLING CODE 7590-01-M

OFFICE OF SCIENCE AND TECHNOLOGY POLICY

White House Science Council Panel on Future Military Technologies; Meeting

Notice is hereby given that the panel named above will meet at 8:30 a.m. on June 22, and June 23, 1982, at Science Applications Inc., La Jolla, California.

The panel will discuss research and development of future military programs.

The meeting will be closed to the public pursuant to 5 U.S.C. 552b(c)(1): All material under discussion is classified defense information. Authority for closing: Director, Office of Science and Technology Policy.

Contact: Dr. Alf L. Andreassen, Office of Science and Technology Policy, 728 Jackson Place, N.W., Washington, D.C. 20500, Phone: (202)395-5684.

Robert D. Linder,

Executive Director, Office of Science and Technology Policy.

June 4, 1982.

[FR Doc. 82-15542 Filed 6-4-82; 10:27 am]

BILLING CODE 3170-01-M

SECURITIES AND EXCHANGE COMMISSION

[Release No. 22521; (70-6558)]

American Electric Power Co., Inc.; Proposal To Issue and Sell Additional Common Stock to Trustee of Employees' Thrift Plan

June 3, 1982.

American Electric Power Company, Inc. (the "Company"), 2 Broadway, New York, New York 10004, a registered holding company, has filed a post-effective amendment to its declaration in this proceeding and an amendment thereto with this Commission pursuant to Sections 6(a), 7 and 12(e) of the Public Utility Holding Company Act of 1935 ("Act") and Rules 50(a)(5), 62 and 65 thereunder.

By order dated April 24, 1981 (HCAR No. 22025) the Company was authorized to issue and sell, from time to time through June 30, 1982 up to 200,000 shares of authorized unissued common stock $6.50 par value to the Huntington National Bank ("Trustee"), as trustee of the Columbus and Southern Ohio

ATTACHMENT A

ATTACHMENT D

TENTATIVE MEETING SCHEDULE FOR THE JUNE 30, 1982 MEETING
OF THE ACRS SUBCOMMITTEE ON THE SYSTEMATIC EVALUATION
PROGRAM - R. E. GINNA NUCLEAR POWER PLANT

8:30 a.m. 1. Introduction (C.P. Siess)

1.1 Purpose
1.2 Goals
1.3 Schedule for Committee Action

8:35 a.m. 2. NRC Staff Introduction (W. Russell/A. Wang)

2.1 When will license conversion be reviewed (approximate time until ACRS review requested)

2.2 How will license conversion be handled. By who?

2.3 Updated Schedule for Remainder of SEP reviews

2.4 Discussion of EDO reply to ACRS Palisades - S.E.P. letter - (who will resolve open issues. What mechanism?)

2.5 If an upgrade is required at an S.E.P. plant under what authority does the Staff require upgrades? What mechanism would be used for an immediately required fix?

9:00 a.m. 3. Introduction by Rochester Gas and Electric Company

3.1 Plant Description

3.2 Operating History

9:10 a.m. 4. Topic List Selection and Resolution (topics to be presented by Staff with comments by RG&E as appropriate).

4.1 List of 21 topics that did not apply to Ginna (5 min.)

4.2 Discussion of 24 topics that were deleted because:

4.2.1 Being considered under TMI-2 Action Plan

4.2.2 Topics being covered by Unresolved Safety Issues (10 min.)

ATTACHMENT B

9:25 a.m. 4.3 Discussion of 58 topics that met current criteria or were acceptable on another defined basis:

4.3.1 List of topics (or issues) that met current criteria

4.3.2 List of topics that were acceptable on another defined basis - type of judgement used by the Staff in decision of acceptability: examples

1. II-4 Geology & Seismology
2. V-6 Reactor Vessel Integrity
3. III-4-B Turbine Missiles
4. Others

10:30 a.m. ****** BREAK ******

10:40 a.m. 4.4 Discussion of 7 topics, plus additional issues that became acceptable as a result of modifications made during the topic reviews

11:20 a.m. 4.5 List of 27 topics considered for Backfit in the Integrated Assessment

4.5.1 List of topics (or issues) not requiring any form of backfitting

4.5.2 Topics (or issues) which will require additional analysis or testing prior to resolution on the need for backfit.

12:00 NOON ****** LUNCH ******

1:00 p.m. 4.5.3 Topics which require (4.5.3.1) physical modification and (4.5.3.2) procedural/administrative changes (list separately)

4.5.4 Topics on which the Staff and Licensee disagree on the resolution

B-2

2:30 p.m.	5.	Discussion of Use of PRA on the Ginna S.E.P. Validity of blending two studies. How did Staff choose to use results?
3:00 p.m.	6.	Discussion by Licsensee on the Worth of the S.E.P./Integrated Assessment as Applied to Ginna
3:30 p.m.	7.	General Discussion on Extension of S.E.P. In light of the Palisades and Ginna Review, should the NRC proceed with Phase III reviews of plants the same age as Palisades, Ginna, and later?
4:00 p.m.	8.	Conclusions
		8.1 Additional questions to licensee
		8.2 Directions to Staff & Licensee for full Committee presentations
4:15 p.m.	*******	ADJOURN *******

ATTACHMENT C

June 30, 1982
Systematic Evaluation Program
Subcommittee Meeting

Attendee List

ACRS

C. Siess, Chairman
W. Kerr, Member
H. Lewis, Member, Part time
J. Ebersole, Member
W. Lipinski, Consultant
I. Catton, Consultant
R. Major, Staff (DFE)
H. Alderman, Staff

NRC Staff

W. T. Russell
C. Grimes
A. Wang
T. Cheng
R. Scholl, Jr.
M. Boyle
R. Zimmerman
J. Lyons

Rochester Gas & Electric

R. Mecredy
G. Wrobel
T. Weis
R. Smith
G. Larizza

NUS

B. Reckman
K. Elliott

LeBoeuf, Lamb, Leiby, & MacRae

H. Voigt
F. Lyman

J. Dunkleberger	New York Energy Office
O. Williams	NuTech
E. Murphy	Westinghouse
R. Snaider	TERA
P. Minson	ARC
J. Preston	ACRS Fellow
W. Baldewicz	ACRS Fellow
J. Nelson	Quadrex
R. Spulak	Sandia Labs
S. Filipoos	ARC
W. Fuller	Wyle Labs
J. McGwen, Jr.	TSI

ATTACHMENT C

Meeting Slides and Handouts for the June 30, 1982 Meeting of the ACRS Subcommittee on the Systematic Evaluation Program (Integrated Plant Safety Assessment of the R. E. Ginna Nuclear Power Plant)

1. NRC Staff slides used by C. Grimes and A. Wang. "SEP Phase II Review - Ginna" Slides 1-60.

2. Slides used by RG&E "History of Ginna Station" - Joe Larizza, Slides 1-4.

3. Slides used by NRC Staff - T. M. Cheng, "SEP Topic II-4, Geology and Seismology, Slides 1-6.

4. Slides used by NRC Staff - M. Boyle, "Turbine Missiles, SEP Topic III-4 B" Slides 1-6.

5. Slides used by NRC Staff - R. Scholl, "SEP Topic V-11.A and V-11.B Requirements for Isolation of High and Low Pressure Systems" Slides 1 - 8

6. Slide used by RG&E - T. Weis, "Deer Creek Watershed Significant Peak Flows Summary of Results" - 1 Slide.

7. Slide used by RG&G - G. Wrobel, "III-3.A Design Basis Groundwater Level" - 1 slide.

8. Slides used by RG&G - G. Wrobel. "Service Water to Containment Fan Coolers and Compartment Coolers and Penetration #101, #110 (Bottom) and #113" - 2 slides

9. Slide used by RG&E - R. Mecredy, "SEP Objectives" - 1 slide.

10. Meeting handout from NRC staff - comments on Ginna integrated plant safety assessment by the NRC staff's Senior Review Group including: J. Hendrie, S. Bush, Z. Zudans, H. Isbin, and R. Budnitz.

11. Memorandum for R. Major, ACRS staff from W. Russell, Chief SEP Branch, Subject: Ginna Integrated Assessment (two .. (1) Topics for which the plant meets current criteria, and (2) topics for which the plant was acceptable on another defined basis, dated June 21, 1982.

ATTACHMENT D

PDR QUALITY CONTROL ACCESSION LIST

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ACCESSION# SN DOC.DATE      FILE LOCATIONS / LEVELS          FILE PACKAGE# DTC    AA1      DIN            DPN DR RA1      SN
 NO.OF PAGES  SIZE                                                          AN1                                RN1
 FORMAL REPORT NO.                                                     AA2  AN2                RA2            RN2
   AVL  DLIST                                                          AA3  AN3                RA3            RN3
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   21pp.      A4       PDR  ADOCK 05000244 0             820630                     *

   PDR
   Summary of ACRS SEP Subcommittee 820630 meeting in          Washington,DC re facility integrated plant safety
   assessment.
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