ACRS-2291
PDR 081985

Aug 29, 1985

TABLE OF CONTENTS
MINUTES OF THE
299TH ACRS MEETING
MARCH 7-9, 1985
WASHINGTON, DC

PROPOSED MINUTES OF THE
299TH ACRS MEETING
MARCH 7-9, 1985
WASHINGTON, D.C.

The 299th meeting of the Advisory Committee on Reactor Safeguards, held at 1717 H Street, N.W., Washington, D.C., was convened by Chairman D. A. Ward at 8:30 a.m., Thursday, March 7, 1985.

[Note: For a list of attendees, see Appendix I.]

Chairman D. A. Ward noted that existence of the published agenda for this meeting, and identified the items to be discussed. He noted that the meeting was being being held in conformance with the Federal Advisory Committee Act and the Government in the Sunshine Act, Public Laws 92-463 and 94-409, respectively. He also noted that a transcript of some of the public portions of the meeting was being taken, and would be available in the NRC's Public Document Room at 1717 H Street, N.W., Washington, D.C.

[Note: Copies of the transcript taken at this meeting are also available for purchase from Ace-Federal Reporters, Inc., 444 North Capitol Street, Washington, D.C. 20001.]

I. Chairman's Report (Open)

[Note: R. F. Fraley was the Designated Federal Official for this portion of the meeting.]

Chairman Ward indicated that an answer from Chairman N. Palladino had been received in response to the ACRS letter of February 11, 1985 appealing the ACRS funding and staffing levels for FY 1985 and 1986. As a result it has been proposed that member as well as consultant travel be curtailed to spread the reduced amounts of travel funds as far as possible. Subcommittee meetings which have been held monthly are to be scaled back to bi-monthly as a cost-cutting measure. It was noted that these matters would be discussed at a meeting of the Procedures Subcommittee on April 9, 1985.

D. A. Ward indicated that some committee members are having difficulty finding the opportunity to review INPO reports under the present ACRS office procedures. R. K. Majors, ACRS Senior Staff Engineer, was asked to obtain additional copies of INPO reports, SER's, SOER's and special reports from AEOD for distribution to those members who are interested in getting them. D. Okrent, D. A. Ward, C. Michelson and C. J. Wylie expressed interest in the reports.

II. Nine Mile Point Nuclear Plant, Unit 2 (Open)

[Note: J. C. McKinley was the Designated Federal Official for this portion of the meeting.]

C. P. Siess explained that Unit 2 at the Nine Mile Point Nuclear Station is a 1080 MWe BWR 5 with a Mark II containment. It is located near the shore of Lake Ontario and shares a site with Nine Mile Point Unit 1 which is a 600 MW BWR that has been operated by Niagara Mohawk Power Company for the last fifteen years. Ajacent to the site is the Fitzpatrick Nuclear plant originally operated by Niagara Mohawk but now operated by the New York Power Authority. He noted that the nuclear steam supply system at Nine Mile Point 2 is very similar to other Mark II's and other BWR 5s and it is almost identical to WPPSS Unit 2 and LaSalle. There are however differences in the plant layout and balance of plant which will be described at this session.

J. C. Ebersole expressed interest in the water cooling system for the three diesels. D. W. Moeller indicated that he had some questions about emergency planning regarding the coordination of Nine Mile Point Unit 2 with Unit 1 and Fitzpatrick. C. Michelson indicated that he wanted to be sure that the issue of isolation of a potential loss of coolant accident outside of primary containment, the HPCI question, is taken up during this session. J. C. Ebersole noted that the plant has simple cantilever downcomers for the safety relief valves (SRV). He wanted to be sure that the Committee discussed the consequences of a SRV downcomer failure.

E. J. Weinkam, NRC Licensing Project Manager, presented the list of outstanding issues on the Nine Mile Point Unit 2 licensing review noting that this plant is not a contested case (see Appendix IV). He explained the significance of the 18 outstanding issues and

indicated that he would review in some detail the issues of Snow Loads, Separation Criteria, Safe and Alternate Shutdown, Essential Lighting and Air Start System. C. Michelson asked if the staff had addressed or considered the differences between Unit 1, a Mark I and Unit 2 a Mark II containment in terms of the interchangeability of operational personnel. Should personnel shift from one unit to the other, possible confusion may result because of the differences between the two units. A. F. Zolnick, Niagara Mohawk Power Corporation noted that that issue would be discussed in the Niagara Mohawk Presentation on Operations Staffing.

E. Weinkam indicated that the Staff in late January 1982 requested that the Applicant justify its reduction in design basis snow load from 85 to 45 pounds per square foot and the reduction in the severe loading conditions. He noted that once the design basis snow loading has been established, the capability of seismic category 1 structures to withstand these loads will be reviewed by the Staff. The Staff expects a submittal on this issue by the end of March.

E. Weinkam explained that Niagara Mohawk has stated that justification by analysis would be used for exceptions to established separation criteria in the power generation control complex cabinets in the control room. The Staff has requested that Niagara Mohawk provide any analysis performed to justify such exceptions to the separation criteria found in IEEE 384 and Regulatory Guide 1.75. The applicant intends to submit a GE analysis by the end of March 1985 to justify

the separation exceptions. J. C. Ebersole took note of the arbitrariness of the separation criteria. He indicatd that since there is a backup control system to assure safe shutdown, the fact that a separation of 6 inches is mandated as opposed to 4 to 7 inches is not a concern.

E. Weinkam reviewed the general criteria for alternate shutdown capability and indicated that the Staff is currently reviewing additional information from the Applicant on this issue. He indicated that the applicant was asked to show how adequate lighting would be maintained for access (ingress and egress) to safety related areas required for safe shutdown for periods longer than 8 hours after the design basis seismic event with attendant loss of off-site power. The Staff assumes nonclass 1E lighting is unavailable following such an event and the Staff will report on the resolution of this issue in a Supplement to the SER.

E. Weinkam explained that the Staff finds proposed diesel generator air start systems at Nine Mile Point Unit 2 unacceptable. These systems would use moisture separators and filters in conjunction with manual blowdown of the air receivers and system piping to preclude air start system contamination and eliminate the need for air dryers. E. F. Zallnick, Niagara Mohawk, indicated that Niagara Mohawk has notified the Staff that they will accede to the Staff's wishes to put in air dryers and will prepare an appropriate submittal.

S. Collins, NRC, discussed Region I's assessment of the construction quality and readiness for operation at the Nine Mile Point Unit 2. He reviewed the history of the NRC's inspection effort concentrating on three SALP reports. He indicated that a SALP Report in early 1984 resulted in an enforcement action and orders issued for both Nine Mile Point Unit 1 and Nine Mile Point Unit 2 in the areas of project management oversite and the implementation of an effective quality assurance program. The Licensee was asked to conduct numerous oversite reviews using a third party. He explained that as a result of oversite inspections focusing on problems found with hardware and a Region I effort to verify the adequacy of both the hardware and the effectiveness of the quality assurance program, a determination has been made that construction of the plant ongoing is adequate and acceptable. He indicated that the Staff still has outstanding concerns regarding a large backlog of approximately 350 items at the beginning of 1985. He noted that as a result of an enforcement action Niagara Mohawk has initiated a hardware verification program, a conservative effort to reinspect hardware at the site.

S. Collins indicated that, during the last SALP assessment of Unit 1, the plant was given a category III rating for operations which, although at an acceptable level of performance, showed a lack of attention to detail in the operation of the plant (see Appendix V). He noted that Unit 1 has a very high capacity factor, and as a result, Region I has confidence in the operational program at Unit 1. The Staff believes that, if the values and initiatives that Niagara Mohawk is undertaking at Unit 1 with regard to their operational program are transferred to Unit 2 and are verified, both plants will be deemed

acceptable. The Committee discussed the relevance of the number of construction deficiency reports as it relates to the adequacy of the quality assurance program and construction quality in general.

C. V. Mangan, Niagara Mohawk V.P. of Nuclear Engineering and Licensing, described the site of the Nine Mile Point Unit 2 facility. He discussed Niagara Mohawk's management philosphy and nuclear experience (see Appendix VI). He mentioned the Corporation's philosphy of voluntary compliance with regard to the process of obtaining a permanent license to replace the provisional operational license for Nine Mile Point Unit 1. He pointed out that, when cracks were discovered in the sensitized safe ends of the recirculation system piping at Unit 1, Niagara Mohawk immediately decided without prompting from the NRC to replace the piping and the safe ends. The same philosphy carries over into the design and future operation of Nine Mile Point Unit 2. J. Ebersole mentioned stainless steel downcomers for the safety relief valves which may be subject to intergranular stress corrosion cracking. The subject was tabled until later in the session. C. Mangan indicated that Niagara Mohawk is very active in industry groups such as the Electric Power Research Institute, INPO, and other organizations concerned with boiling water reactors.

C. V. Mangan explained that Management Analysis Company (MAC) was hired by Niagara management in late 1983. The engaging of MAC was considered preventive and prudent with the goal of ensuring the timely completion of the project. Toward the completion of construction a gradual phase-out of MAC employees will begin with most of the positions and functions now handled by MAC given to Niagara Mohawk personnel now being trained. W. Kerr thought it inconsistent that an organization such as MAC was called in to deal with problems related to construction but will not be involved in the operation of the new plant. He concluded that MAC may have been brought in for cosmetic purposes to deal with more stringent QA regulatory requirements. C. V. Mangan agreed that the problems were basically quality-related - construction management and the quality organization. C. P. Siess suggested that QA could not keep up with the construction on Unit 2 and there were stop work orders until Niagara Mohawk could catch up with resolution of construction deficiency reports. He suggested that this was the reason MAC was brought in to assist. C. V. Mangan did not agree that QA could not keep up with construction but did agree that MAC was hired officially after Region I expressed its concerns regarding QA problems at the Nine Mile Point site.

G. A. Reed asked if Niagara Mohawk is considering using hydrogen addition to the primary water chemistry, in addition to piping material improvements, to control stress corrosion cracking in both of its boiling water reactors. C. V. Mangan indicated that Niagara Mohawk is in the process of evaluating hydrogen addition for both Units 1 and 2. There are some potential problems with radiological dose rates and the generation of hydrogen in the gaseous waste stream. These matters required further evaluation. D. W. Moeller asked if Niagara Mohawk had estimated the amount of dose increases that could be expected with hydrogen water chemistry. C. V. Mangan indicated

that it was of the order of 2 to 5 times. W. G. Hooten, Niagara Mohawk Executive Director of Nuclear Operations, discussed the capabilities of the nuclear organization at Niagara Mohawk (See Appendix VII). D. Okrent asked regarding the level of competence in the control room for beyond design basis accident control. J. C. Ebersole postulated a long term AC power outage and asked how the operators would deal with the situation of high containment pressure relief during such an accident. R. B. Abbott, Nine Mile Point Unit 2 plant's supervisor, indicated that the emergency operating procedures for this specific event address a venting action of the primary containment should the pressure get to around 50 pounds.

C. Michelson expressed concern that the SALP deficiencies regarding lack of attention to detail in the operation of Unit 1 should feed through to Unit 2 so that Unit 2 will suffer from the same problem. W. G. Hooten explained that MAC was contracted to assist in corrective actions on Unit 1 and to see that the same deficiencies do not feed through to Unit 2 even though there will be some overlap in personnel practice, policies, and procedures. C. Michelson asked for definitive corrective actions that have been taken. W. G. Hooten sited Niagara Mohawk Training Programs which are strongly concerned with the maintenance of the proficiency of the reactor operators. R. Abbott added that special training programs for technicians and operators on primary containment integrity are being conducted. Efforts are being made to more effectively involve personnel in surveillance procedures tests on equipment (such as main steamline radiation monitors).

J. Perry, Niagara Mohawk Director of Quality Assurance, discussed the quality assurance program at the Nine Mile Point Unit 2 plant (see Appendix VIII). As an indication of the increased level of attention Niagara Mohawk has been putting in the area of quality assurance, he pointed to the ratio of quality assurance/quality control personnel to site total manual labor personnel of 1 to 12 in late 1983 and 1 QA/QC person per 8 labor personnel in late 1984. C. P. Siess asked about the ratio of QA personnel to QC personnel. J. Perry indicated that it is about 1 to 5. The Committee discussed the quality assurance deficiencies found by the CAT team report in January 1984. W. Kerr reminded the Committee of Niagara Mohawk's voluntarism and initiative. He pointed out that it appears in this situation that Niagara Mohawk was aware of the deficiencies, but that the NRC had to meet with management to encourage attention to these deficiencies. J. Perry did admit that the CAT inspection got Niagara Mohawk Management's attention and heightened interest in management involvement. J. C. Ebersole asked about the status of Nine Mile Point Unit 1 regarding Appendix A and Appendix R requirements for fire protection. N. Rademacher, Niagara Mohawk, pointed to a recently completed successful audit by the NRC Staff of Niagara Mohawk Appendix R fixes. He indicated that Niagara Mohawk was the second utility in the United States to get through an Appendix R audit without a violation.

D. Okrent asked if the quality assurance effort has been factored into the preoperational test program for Unit 2. J. Perry explained that the construction QA operation and the startup and test QA operation are quite different in that the startup and test activities are

conducted by the project, run by Niagara Mohawk, and not a contractor. The organization running these tests includes personnel from the operating plant (Unit 1) with hands-on experience. A separate quality assurance group dedicated to Unit 2 will handle the startup and test QA activities in terms of reviewing both administrative as well as test and surveillance procedures. He explained Niagara Mohawk efforts to reinspect (with current design requirements in mind) using engineering evaluations of deficiencies that were found. Niagara Mohawk has concluded that the end products will perform their intended function. The problems with quality experienced at Nine Mile 2 are not analogous to those experienced at the Zimmer plant. He mentioned a Quality First Program to address situations where plant personnel come forward with a quality concern. A team of qualified quality assurance individuals has been assigned to investigate the facts and conclude whether the allegations have merit.

R. B. Abbott discussed operations, staffing, and training (see Appendix IX). He indicated that Nine Mile Point's training program includes extensive training for nonlicensed personnel to support the Nine Mile Point stations. The nonlicensed operator training ensures eligibility to enter the licensed operator training program. Licensed operator training programs stress the necessary knowledge and skills required for successful licensing. A plant reference simulator was available for the first cold license training sessions. He pointed out that the Nine Mile Unit 2 simulator is designed as a plant reference simulator that meets the requirements of ANSI-3581 and Regulatory Guide 1.149. Its features include the ability to freeze the action, run in slow time, or run in fast time. F. J. Remick asked how many upset conditions or malfunctions are programmed in this simulator. R. B. Abbott indicated that there were over 100 scenarios. F. J. Remick asked if all of the Shift Tenchical Advisors are licensed as SRO's. R. B. Abbott indicated that they are all licensed. F. J. Remick asked whether the option of a separate STA or combined SRO/STA would be used on Unit 2. R. B. Abbott indicated that Niagara Mohawk intends to have a mix of qualified SRO's to fill the STA functions. But, if given the choice right now, Niagara Mohawk would stick with the separate STA. F. J. Remick asked how many crews are planned on Unit 2. R. B. Abbott indicated that there would be 8 operating crews and even if one were eventually dropped, there would never be less than 7 operating crews. It was noted that there was a five man dedicated fire protection team assigned to each crew and that each crew consisted of about 20 persons.

F. J. Remick asked if Niagara Mohawk expects a morale problem from the NRC conduct of requalification examinations at Unit 1. R. B. Abbott indicated that there is a general apprehension on the part of the operators, but he did not expect major problems. F. J. Remick asked how much time during the requalification program Unit 2 operators will spend on the simulator. R. B. Abbott indicated a minimum of 2 to 3 weeks. F. J. Remick noted that Niagara Mohawk intends to license reactor operators on Unit 1, though it intends to use them on Unit 2. He asked about the possible confusion for the operator between Unit 1 and Unit 2. R. B. Abbott indicated that the licensing process on Unit 1 would give the operator an appreciation of nuclear plant operation

and make him more easily trainable for Unit 2. G. A. Reed asked if Niagara Mohawk uses testing for natural aptitude or ability in its selection of mechanics or operators. R. B. Abbott indicated that only technicians are given exams during the course of preemployment interviews. Other new hires are selected on the basis of documented qualifications and an interview. He did indicate that Niagara Mohawk was aware of the Edison Electric Institute tests that G. A. Reed was citing.

J. C. Ebersole raised a question about a semi-automatic blowdown through the safety relief valves and how it would be addressed as an operational problem. The Committee briefly discussed the issue. D. Okrent raised the issue of a loss of all offsite power incident. He speculated that safety relief valves were stuck open for about 3 hours into the event and the diesels, previously unavailable, were now ready to be started. He asked what sequence of operation would be used to restore the Unit and who would be in charge of the procedure. N. Rademacher, Niagara Mohawk, indicated that the station blackout study which is basically this scenario was not yet complete. D. Okrent explained that he was considering stuck open relief valves in addition to a station blackout which would amount to a station blackout with a steam line rupture. His concern related to a severely degraded core. Has research been completed that would indicate it would be better to use core sprays or flooding and what would this do to the source term? Rademacher noted that regardless of any additional failures such as stuck open relief valves, Niagara Mohawk could provide little more than a preliminary estimate of various ramifications of this scenario such as the time to core uncovery. R. B. Abbott indicated that the shift superintendant would make the decision. G. A. Reed asked if there is a duty on-call individual with engineering expertise. R. B. Abbott indicated that there was.

C. D. Terry, Niagara Mohawk Manager Nuclear Engineering, described certain enhancements of Nine Mile Point Unit 2's decay heat removal systems which are somewhat different than at other BWRs. He mentioned that enhancements have been implemented in the reactor building design and equipment location which contribute to improved overall maintainability and reliability of systems (see Appendix X). He also pointed out enhancement of the traditional Mark II containment design, including the addition of north and south auxiliary bays to the reactor building to relieve congestion typically found inside most facilities. They expect the changes to enhance the operatibility and reliability of RHR and ECCS equipment by permitting distinct isolation compartments. C. Michelson asked if each compartment is designed for a potential steamline pipe break with hot steam filling compartment(s) which fails the compartment(s). C. Terry indicated that there is sufficient blowout capability within the compartment to allow for a potential pipe break. C. Michelson asked what would ultimately happen if there were an unisolated steam line failure of the RCIC system. He indicated his concern was the kind of pressurization that would occur and the venting that would be required if the valves do not close. N. Rademacher was not able to confirm that the valves would close under dynamic conditions. He indicated that Niagara Mohawk would look into this matter. C. Michelson discussed a case of an RHR suction line

break with suppression pool water flooding compartments. C. Terry presented a diagram of the actual arrangement of the flood troughs which allow water to spill over into the compartments (see Appendix XII). In addition the primary containment diameter was increased by two feet which increases the containment volume, decreases congestion, and increases available suppression pool water inventory. The suppression pool is lined with stainless steel to improve the cleanliness of the water and reduce degradation of the pool itself.

E. R. Klein, Niagara Mohawk, described three unique features of a hydrodynamic character for the Nine Mile 2 Mark II containment (See Appendix XII).

- Self supporting dry well floor
- Vacuum breaker modifications
- Downcomer design without use of bracing

C. P. Siess asked regarding bypass leakage testing from the dry well. N. Rademacher indicated that one creates a differential pressure between the dry well and the wet well and measures the leakage. The Committee discussed the downcomer design and its characteristics regarding stresses and resonance. D. Okrent asked if the Staff had audited the design of the downcomers. J. Langton, NRC, indicated that the Staff is currently looking at downcomer loads. It is a confirmatory item for Nine Mile Unit 2 (Pool Dynamics). In answer to a question by D. Okrent, R. Bosnak, NRC, Division of Engineering, indicated that the downcomer lines are Class 2 and would require inservice inspection.

N. Rademacher explained that in response to NRC requirements in 10 CFR 50.62 to mitigate the consequences of an ATWS, Nine Mile Unit 2 has installed an alternate rod insertion subsystem, a recirculation pump trip subsystem, and an automatic-initiated standby liquid control system to provide 13 weight percent boron solution to the core (see Appendix XIII). W. Kerr asked the Applicant for the number used for the likelihood of failure on demand of the pump trip logic in their quantitative analysis of scram reliability. N. Rademacher was unable to produce a backup slide with that information and it was assumed that it would be provided at a later time.

N. Rademacher described operating personnel experience, improved equipment, and enhanced quality programs to effect a reduction in the number of scrams. He described control rod drive hydraulic system changes including material upgrades for intergranular stress corrosion cracking (upgrades to the low carbon stainless steel). C. P. Siess asked if the hardware modifications contemplated for Unit 2 are on Unit 1. N. Rademacher indicated that Nine Mile Unit 1 has alternate rod insertion and also recirculation pump trip. C. Michelson asked about safety grade provisions to prevent overfill of the reactor vessel. N. Rademacher indicated that Niagara Mohawk does not have a safety grade interrupt capability.

N. Rademacher indicated that the fire protection program at Unit 2 is essentially the same one implemented at Unit 1 which uses the defense-in-depth concept (see Appendix XIV). He indicated that the fire hazards analysis for Unit 2 has been completed and submitted to the Staff. It is now an open item still in the process of review. C. Michelson asked how Niagara Mohawk addresses the question of the potential for inadvertent actuation of water sprays or deluge systems. N. Rademacher indicated that two separate actuations have to occur, a signal from the fire protection system would cause a valve to open admitting water to the dry pipe system and then thermal links on individual nozzles would have to release. C. J. Wylie asked about fire protection in the diesel generator rooms. N. Rademacher indicated that the same pre-actuation sprinkler system as previously described is used, not a water spray system.

C. Michelson asked if CO_2 is used anywhere. N. Rademacher indicated that it is used in three switchgear rooms in the control building. C. Michelson asked how Niagara Mohawk prevents spurious operation of the CO_2 systems caused by common cause events such as those of seismic origin. N. Rademacher indicated that there are no safety-related components in the fire system. Reliance is basically on redundant valves from the tank to control the flow into the switchgear rooms with manual operation by the fire brigade. He indicated that there is automatic initiation of CO_2 on appropriate signals. C. Michelson asked how Niagara Mohawk guarantees against actuation of CO_2 systems during an earthquake if the systems are not seismically qualified. N. Rademacher indicated basic reliance on manual shutoff of the system. C. Michelson asked if Niagara Mohawk claims that the safety related equipment in the diesel generator rooms will continue to function properly in the presence of carbon dioxide. C. D. Terry asserted that the equipment is designed to run with CO_2 in the room. C. Michelson pointed out that Niagara Mohawk cannot cite any specific testing of the equipment under those conditions and he remained unconvinced that the equipment would function under those conditions. D. A. Ward asked if the pre-actuation system enabling valve is triggered by a temperature sensor or a smoke sensor. M. Cameron, Niagara Mohawk, indicated that the detectors in the CO_2 activation areas are ionization detectors. There are thermal detectors on the pre-actuation sprinkler systems. Cross zone ionization detectors are utilized on the thermal flood CO_2 systems. C. Michelson noted that there is almost complete assurance that the seismic event that creates dust coming out of wall cracks will be picked up by the ionization detectors, thought to be smoke, and actuate CO_2 detectors. He suggested that the applicant investigate this matter.

N. Rademacher discussed electrical separation (See Appendix XIV). He explained that the reactor protection system has four separate tunnels for redundant divisions. J. C. Ebersole asked if Niagara Mohawk carries out a hypothetical test of a transient fire from room to room. N. Rademacher agreed that Niagara Mohawk does assume the loss of areas where a fire occurs and does a fire hazards analysis simulating someone carrying flammables from area to area. The objective of the study would be to see whether the plant could be safely shut down using one of the other tunnels or other switchgear. J. C. Ebersole

asked if there is dependence on fire damper functioning which is notoriously unreliable for containment of a fire in a particular region. N. Rademacher indicated that heat links are used in the dampers to close them. C. P. Siess asked if there are dampers between reactor protection system tunnels. N. Rademacher indicated that there is a separate HVAC system for each tunnel and no need for dynamic closures.

D. Pike, Niagara Mohawk, discussed the NRC requirements for a safety parameter display system (SPDS) as set forth in NUREG 0737 (see Appendix XV). He indicated that this system features a combined Quadrex/Honeywell computer-based system which provides control of radwaste facilities as well as generator temperature monitoring. He discussed a portion of the system called the Emergency Response Facility System (ERFS) which is composed of a data base of about 552 analog and digital points functional in the control room, the technical support center, and the emergency offsite facility. The ERFS has the capability of forming about 180 graphic displays creating bar charts, XY plots for the trending of variables, alarm summaries, and various group logs. It also has a historical event retrieval feature which is said to identify various trigger signals such as LOCA, scram, and various transient signals which will trigger display of historical event data saved on memory. J. C. Mark asked if the system is safety grade, in the sense of being single failure proof or seismic resistant. D. Pike indicated that it is neither seismic nor safety grade. It was noted that there are redundant computers to assure the required reliability.

D. A. Ward asked if the emergency operating procedures (EOP) provide for using the SPDS. R. B. Abbott indicated that the EOPs addressed control panel instrumentation. Nine Mile Two operators will not rely on the SPDS for responding to accident transient conditions. D. Pike added that the SPDS is also available to the shift technical advisor to utilize for analysis purposes while the operators concentrate on handling accident situations. D. Pike explained that the SPDS is a Red/Green system where red is "bad" and green is "good." Red is used to indicate off-normal conditions, R. Axtmann pointed out that about 20 percent of the human population is red/green color blind. He asked if this affects Niagara Mohawk's employment policies. D. Pike indicated that irrespective of the fact that the NRC has no regulation regarding this matter, Niagara Mohawk has decided that it will not hire color blind reactor operators.

D. Pike indicated that the remote shutdown capability at Nine Mile Point Unit 2 meets NRC requirements and the remote panels fulfill all Appendix R requirements when coupled with some local operations. He described the physical layout of the rooms in which the remote shutdown system is located. He indicated that the two panels are separated by a three hour fire wall and a three hour rated door. C. Wylie asked if there would be any effect on the operation of the remote shutdown system from an exposure fire in the control room which shorts out all of the boards. D. Pike indicated that there are transfers on the remote shutdown panels to isolate the circuits. He explained that the remote shutdown rooms are located two elevations

below the control room and are electrically separated from the control room by the use of key lock transfer switches. Each room has its own dedicated safety related HVAC system.

C. Michelson noted that Nine Mile Point 2 has redundant backup control panels and can get along with only one of these panels. He asked if Niagara Mohawk had a system that negates undesirable functions that are derived from hot shorts in the control room's circuits that blow fuses before transfer to the remote shutdown panel can be made. D. Pike indicated that when one throws the transfer switch a redundant set of protected fuses will allow operation of pieces of equipment. J. C. Ebersole asked about negating functions that cannot be tolerated such as valves being opened when they should be closed. D. Pike indicated that in addition to the panels there are disconnects to satisfy Appendix R requirements which disconnect those circuits from the control room. J. C. Ebersole asked if an isolated fire in one of the remote shutdown rooms could disable equivalent equipment in the main control room. D. Pike indicated this is possible but the redundant remote shutdown equipment would still be functional.

N. Rademacher discussed an analysis performed on the reliability for ATWS (see Appendix XVI). He indicated that a PRA analysis was done for the Nine Mile Point Unit 2 for core melt accidents. This analysis identified sequences which contribute to risk (see Appendix XVII). He noted that although external events were not evaluated in the PRA, the analysis does include the loss of offsite power as an accident initiator. The PRA also includes component level analysis for plant systems and various containment failure modes. The study includes an evaluation of risks and socioeconomic impacts. J. C. Ebersole noted that while Niagara Mohawk claims three tracks for onsite AC power, including a third diesel, there is a curious dependence of one of the tracks on the other two regarding cooling water to the diesels. Cooling water is provided to the third diesel from the output of either of the other two with the result that there appear to be really only two divisions of onsite AC power. He suggested that this configuration might be contradictory to General Electric's interfacing requirements. N. Rademacher indicated that it was a deliberate design to address all situations for LOCA and lesser transients. A study is now underway to address the situation of the third diesel generator depending on the two division service water system for engine cooling. Results are expected later this year.

J. C. Ebersole noted that Nine Mile Point has the unusual feature of having isolation valves on each side of the recirculation pumps. During operating modes for which the isolation valves on each side of the recirculation pump are closed, he asked by what mechanism Niagara Mohawk assures against a pump start. He asked if this mechanism is redundant and safety grade and what might happen if the pump started and ran for an extended period. N. Rademacher offered to check on the matter and provide an answer to the questions at a later date.

D. W. Moeller noted that the control room has two air intakes but a radiation monitor located in a common header where the two paths meet. He thought that this was not an optimum situation for detecting

accidental airborne releases into the control room. D. Pike indicated that the detection of radiation from either train will cause the interception by special filter trays to filter the radiation (particulate and charcoal filters). M. Stockman, Stone and Webster, explained that, while there are two separate air intakes, they are not considered remote since a remote intake according to NRC requirements requires 1,500 feet of separation. These intakes are just the width of the control room, about 100 feet. Therefore, there is no choice for the operator regarding one air intake as opposed to another. The result is a 50 percent dilution.

D. W. Moeller asked if Niagara Mohawk considers the possible impact of an earthquake on offsite emergency response. P. Volza, Niagara Mohawk, explained that it is not required for Niagara Mohawk to look at an earthquake inasmuch as it is a very low probability event at the Nine Mile Point site. He indicated that emergency planning takes into consideration adverse weather conditions typical of the area, such as severe snow and thunderstorms. D. W. Moeller asked if Niagara Mohawk had looked at the potential impact of an earthquake on evacuation of the public. P. Volza indicated that Niagara Mohawk did not look at an earthquake at all with regard to offsite emergency planning.

C. Michelson asked to what extent Niagara Mohawk applies QA and seismic requirements to the reactor water cleanup system outboard of its isolation valves. N. Rademacher indicated that the entire system back to the reactor is seismically designed but is not QA Class 1. C. Michelson asked about provisions for venting of steam in the unlikely event that Niagara Mohawk experiences a break in the reactor water cleanup system under the conditions of getting isolation or not being able to isolate the break. N. Rademacher explained that the compartment walls for rooms housing the RCIC are designed to take a continuous blowdown. The flow would go through blowout panels out to the upper elevations of the reactor building (a relief path). C. Michelson pointed out that a break in the reactor water cleanup system is really equivalent to a recirculation line break if you do not isolate it because the line comes from the recirculation system. He pointed out that the fluid coming from the break is about two-thirds liquid phase and one-third vapor phase. The vapor phase would be vented to the atmosphere. He asked what assurance there is that the liquid phase does not interfere with operation of engineered safety features designed to keep the water at two-thirds core level inside the reactor vessel. D. Pike indicated that Niagara Mohawk has not specifically addressed this issue for these particular valves but will do so in the future.

D. Okrent asked the NRC Staff if the Nine Mile Point site had been evaluated for the safe shutdown earthquake using the Lawrence Livermore expert opinion approach. E. R. Klein, Niagara Mohawk, briefly discussed the seismic evaluation for Nine Mile Point (See Appendix XVIII). D. Okrent asked if an examination had been made of the significance of seismically-induced relay chatter at Nine Mile 2. D. Pike explained that Niagara Mohawk seismic testing of instruments mounted in electrical panels does have the circuits monitored for

relay chatter. The occurrence of relay chatter and an evaluation of its significance would be documented.

D. Okrent noted that recent PRAs have considered the vulnerability of nuclear plants to earthquakes two or three times greater than the SSE. He asked if Niagara Mohawk had determined if the Nine Mile Plants would be vulnerable at that level. D. Pike indicated that Niagara Mohawk had not specifically addressed this issue although there is a certain amount of conservatism and added margins built into the plant. D. Okrent asked if Niagara Mohawk conducted a walkdown regarding systems interactions. C. Terry explained that Niagara Mohawk did not conduct a single walkdown specifically for the purpose of looking for systems interactions but did conduct a series of walkdowns that were directly related to that type of evaluation. Of particular interest at Nine Mile 2 were seismically-induced interactions associated with nonsafety-related equipment suspended over safety-related equipment. D. Okrent asked if a systematic comprehensive look was done regarding the possibility of internal flooding. C. Terry indicated that a specific evaluation of that was done.

J. C. Ebersole noted that boiling water reactors discharge waste heat into the cooling water predominantly by a single wall heat exchanger in which the secondary fluid could be raw river water in this case. He noted that this is a predominant method of heat rejection. He asked if there was sufficient sensitivity to detect radiation leaks into the raw water for normal operating conditions where the primary coolant contamination level was nominal so that one could detect a serious accident which might involve both heat exchangers leaking at the same time. This assumes that there is radiation detection on the secondary side of the heat exchangers. N. Rademacher noted that there is a radiation monitor in the flow stream and he thought that there was sufficient sensitivity to monitor the running contamination level.

C. Michelson asked regarding the water cleanup system whether the staff had looked at the possibility of pipe breaks outside the containment for this particular plant and the environmental qualification of equipment in the lower portions of the building which might see the hot water issuing from the break. R. LaGrange, NRC Equipment Qualification Branch, indicated that the Staff is still awaiting information regarding this Applicant's environmental qualifications program and has not yet begun its review.

The Committee decided that it would be able to write a letter for an operating license on Nine Mile Point Unit 2 plant.

III. Recent Events at Nuclear Facilities. (Open)

[H. Alderman was the Designated Federal Official for this portion of the meeting]

A. HPCI, RCIC Inoperability at Hatch 1

G. Rivenback, NRR, indicated that an event occurred on January 16, 1985 during a power escalation following a long outage for

refueling at the Hatch plant. Loss of a vital AC bus at 60 percent power caused loss of feed water control and a reactor vessel water level trip on low level. The HPCI and RCIC both came on automatically and began to refill the vessel. When the primary supply to that AC bus was turned back on and feed water pump "A" regained full level output, the HPCI and RCIC and reactive turbines tripped on high level. In trying to reset the HPCI and RCIC for future operations, the HPCI turbine stop valve was stuck in the half open position and could not be reset (see Appendix XIX). J. C. Ebersole asked for comment on the implications of this event noting the loss of all emergency high pressure feedwater. E. Rossi, NRC, indicated that there have been a number of problems recently with the reliability of the HPCI and RCIC systems a number of which involved deficiencies in maintenance. J. C. Ebersole agreed that it ought to be noted as a maintenance problem which should be folded into the ACRS' maintenance activities. G. A. Reed indicated that the Maintenance Subcommittee would certainly add this to its list of issues regarding problems with valve reliability.

B. RHR Pump Valve Failure at Browns Ferry 1 and 2

W. Long, NRR, discussed the failure on January 30, 1985 of some RHR system valves to operate, after a vibration test was performed on the Unit 3 RHR pump. The valve failed to open while the plant was operating at power and the RHR system was lined-up in the suppression pool cooling mode in which water is taken out of the suppression pool, pumped through the heat exchangers for cooling and then returned through that valve back to the pool. It appeared that flow-induced vibration during RHR torus cooling mode operation caused loosening and breakage of end bell bolts on the valve operator motor causing the end valve to fall off. The valve became electrically inoperable and failed in the open position. The normal position is closed.

W. Long mentioned the failure of an RHR suction cross tie valve which occurred on January 30 during the refueling outage at Browns Ferry Unit 2. The valve would not operate electrically and an electrical problem existed at the local control station. It was found that the valve became inoperable in the electrical mode due to corroded terminal blocks and switches at the local control station. Grounding at the local control station caused fuses to blow preventing electrical operation. The valve was also found to be inoperable mechanically in the hand wheel mode. He indicated that the cause of the electrical failure appears to be condensation in the junction box and the hand switch which are both supposed to be sealed. C. Michelson suggested that both of these items are required to be environmentally qualified. He asked why they failed. W. Long indicated that they had not been properly sealed. It was noted that there were no safety functions lost as a result of either valve failure.

W. Long indicated that with the failure of the RHR valve on Browns Ferry 3 and the failure of this RHR suction cross tie

valve on Unit 2, Unit 1 was put into a limiting condition for operation. TVA has requested technical specification amendments to delete these limiting conditions of operation and the associated surveillance requirements so that shutdown of an operating unit could be avoided when work is to be done on such a valve maintenance problem. C. Michelson asked that the Staff provide the Committee a copy of the report on the RHR valve which failed due to flow-induced vibration so that the ACRS Subcommittee on Reliability Assurance can review it.

C. Inoperability of Both Safety Injection Pumps on Catawba Unit 1.

E. Weiss, NRC, explained that an event occurred at Catawba Unit 1 on March 5, 1985 when the plant was in the startup mode at zero percent power. This event involved a case of both safety injection pumps being inoperable. The probable cause was not only inadequate operator training but inadequate operator understanding of the relationship between testing of the solid state protection system and the availability of the safety injection pump.

E. Weiss explained that pump 1B was taken out of service for maintenance at 7:45 a.m. and was tagged-out in that condition until 2:00 p.m. At 9:20 a.m. the shift supervisor authorized the bimonthly testing of train A of the solid state protection system which lasted to about 10:30 a.m. During that time the safety injection pump on train A was inoperable. The result was an overlap of about 26 minutes when neither pump was actually available to deliver water should a LOCA have occurred. J. C. Ebersole pointed out that a major overhaul of the technical specifications is now contemplated which he hoped would move toward a so-called matrix configuration which would preclude simultaneous outages but which would reflect both plant conditions and the fact that one can disable plants from a whole host of possible sources. E. Rossi anticipated that that would not be in the technical specifications. There would be a simplification of the language in the technical specifications, and any so-called matrix configuration would probably be in either what are called supplemental specifications if the technical specifications are divided into two parts, or it would be included in control procedures.

D. Malfunction of the Reactor Protection System at Sequoyah Unit 2

E. Weiss explained that the Sequoyah Unit 2 had a reactor protection system malfunction at 96 percent power on January 12, 1985, when the reactor tripped and breaker A failed to open automatically upon a valid reactor trip signal. He noted that inadequate maintenance procedures and post-maintenance testing caused this event. He explained from a diagram of Westinghouse's UV output circuit on the reactor protection system that an instrument technician shorted and burned out a transistor when measuring voltage across the UV trip coil. He noted that all Westinghouse plants designed using the solid state protection

system are susceptible to this type of failure. North Anna Unit 2 experienced three similar failures except that these failures occurred during a plant outage and were detected and repaired prior to returning the plant to power. W. Kerr suggested that this condition must violate GDC 19 which calls for fail safe circuitry. E. Rossi pointed out that the reactor protection system is designed to be fail-safe for certain kinds of problems, one of which is loss of power. He pointed out that the GDC does not say that everything has to be fail safe. The terminology is "fail-safe for certain items such as loss of power" where feasible. W. Kerr suggested that it would certainly be feasible to eliminate this fault with a slight redesign in the circuitry. He saw no reason why, in testing the circuit such as this one, one could so easily disable a transistor. The Committee discussed the vulnerability of the UV output circuitry. E. Rossi pointed out that the real lesson is when you have two trains of identical equipment, solid state circuitry, you can damage the circuitry by maintenance and that can then become a precursor to a common mode type failure. H. W. Lewis thought the circuit poorly designed since it was wired in such a manner that the transistor could damage the rest of the circuit when burned out.

J. T. Beard, NRC, noted that when this first occurred at a reactor in Brazil, Westinghouse undertook a review of this particular circuit design and freely admitted that it could be made better. They concluded, however, that the design in its present state was acceptable and met all the regulations. It was therefore not necessary for it to be changed. E. Rossi indicated that the regulatory criteria are only that the system meet the single failure criterion and since there are two redundant solid state protection systems, no matter what happens to one, there is not a problem unless the other system goes out simultaneously. P. Docherty, Westinghouse, explained to W. Kerr that since the probability of incidents of the failure of the transistor was low even though it would be a simple repair on the circuit, Westinghouse chose not to correct the vulnerability. W. Kerr said he was not impressed with that logic, even though he noted that it prevails in the industry at large.

E. Failure of Containment Tendon Field Anchors at J. M. Farley Unit 2

G. Lear, NRR, indicated that the failure of containment tendon field anchors at the J. M. Farley Unit 2 was reported to the NRC on January 28, 1985. D. Okrent asked for a summary of the generic significance of this item if any. G. Lear indicated that the implications and a resolution of the generic aspects of the problem are still under study. An analysis of some anchor heads that were cracked and fractured has shown that it was hydrogen induced grain separation which initiated cracks on the surface of the inside of the anchor plates which, under stress or load, subsequently cracked. The source of hydrogen was deemed to be water, a corrosion cell identified between the zinc which was the inside surface of the end caps and the cable sheaths which run

along the path of the tendons. There was ionic exchange hydrogen generated against the intergranular action due to the hydrogen itself with the heavy stress imposed. He noted that the fact that water got into the containment sheaths is a generic problem. He acknowledged the need for the NRC to determine whether long term surveillance requirements on all operating plants are necessary. He did point out that this is a long-term slow process which will not require precipitous, calamitous, action. Instead it requires an orderly approach to a program of resolution by means of surveillance and perhaps some action items. E. Rossi indicated that two IE information notices have been issued on this matter and the Staff is still looking at the need for further action to address the generic implications. G. Lear indicated that from a safety point of view, there is much margin in the structural design and at this time there does not appear to be any safety implications.

IV. GESSAR II (Open)

[Note: R. K. Major and R. Savio were Designated Federal Officials for this portion of the meeting]

D. Okrent indicated that this is the first meeting at which the Committee has discussed a forward looking final design approval (FDA) for the General Electric (GE), GESSAR II design. He noted that there have been three Subcommittee meetings since the last full Committee review on GESSAR II and at least two more are planned in the near future. He expressed concern regarding how the NRC Staff would use the PRA developed by the Applicant in reviewing the request for the final design approval in view of all the uncertainties. Of particular interest will be how generic issues and unresolved safety issues are brought to closure. He noted that GE has submitted a PRA which includes a section on external events as well as internal events. There have been some questions by ACRS consultants on the internal event portion of the PRA. The ACRS consultants have been asked to review the external event portion of the PRA and will provide comments in the near future.

D. Okrent noted that GE obtained certain results in their PRA which differ from those calculated by the Staff and their contractors. These differences are largest on the issue of seismic events, the principal external event examined. GE's best estimate calculations for modifications considering the cost of equipment are substantially different from cost results derived by some of the Staff's consultants. Essentially all of the modifications evaluated by GE were found not to be cost beneficial. This will be one of the principal subcommittee topics at the next meeting of the GESSAR Subcommittee. The question of measures to design in to reduce the probability of sabotage will be another topic at the next subcommittee meeting. J. C. Ebersole asked if the Committee ought to continue to look hard at mitigation capability for core melt as the probability of core melt approaches a very low number with the continual addition of prevention features. D. Okrent pointed out that the Committee, in a past letter considering severe accident policy, indicated that one

should not depart from the defense-in-depth concept. He suggested that it would be helpful to have some containment performance criteria. He noted that the British and Italians are looking for fairly good containment performance even in the presence of core melt.

B. Wolf, GE, indicated that the current severe accident review of the BWR 6 Mark III design establishes a significant milestone. It is the first to undergo a comprehensive severe accident review. He also indicated that the plant design under discussion has gone further than any design in the past in meeting the intent and spirit of goals enunciated by the Committee, the Commission, the Staff and the Congress itself. The goals call for a single party such as a vendor to be responsible for the total plant design, for the plant to be fully designed when licensed, and for the design to be safe. He indicated that GE will not enter a rulemaking after the granting of the FDA because there does not appear to be a market in the United States for nuclear plants. It does not make much sense to go through a protracted rulemaking hearing well before the resumption of nuclear power orders for at least two reasons.

- Legislation before the Congress may impact on rulemaking proceedings
- New knowledge may become available so that the design can be simplified

B. Wolf suggested that if some adverse knowledge comes forward in the interim, GE would be able to modify the design appropriately and go into the rulemaking closer to the time when nuclear power will again be needed. He was asked if GE was confident that the design would endure and not be subjected to a large amount of backfitting. B. Wolf indicated that he did not anticipate a large amount of backfitting. He indicated that one goal is to be able to construct a plant in five years. He noted that this approach would be similar to that taken with the Japanese standardized plant.

C. Thomas, NRC, listed requirements for the forward-looking FDA. Design approval would involve meeting those requirements currently proposed in the severe accident policy statement. Approval of the Severe Accident Policy Statement by the Commission is expected.

D. Scaletti, NRC Project Manager for GESSAR, presented an historical chronology of the severe accident review for GESSAR. He noted that the Staff has issued supplements to the SER dealing with severe accidents, internal and some external events (PRA) excluding seismic events, unresolved safety issues and medium and high priority generic issues, compliance with the CP&L rule, and compliance with the standard review plan rule. C. Michelson pointed out that there are a number of generic safety issues that have not yet been prioritized. He asked if the Staff will wait until prioritization has been completed before reviewing these issues as they apply to GESSAR. C. Thomas indicated that the Staff is of the opinion that there will continually be new issues identified that will have to go through the prioritization process. Therefore, the Staff has taken the position

that they will only consider those issues that have been prioritized. Any issue that has been designated a USI or medium or high priority generic issue during the review process would be considered.

R. Bernero indicated that Commission approval of severe accident policy may not be imminent. He explained that a severe accident review of a standard plant as set forth in the severe accident policy statement is not exclusively a PRA review. It is an ordinary review along three fronts:

- Classic deterministic review or standard review plan review
- Review for resolution of unresolved safety issues
- Severe accident PRA review in which PRA techniques are used to find unique vulnerabilities and enhancements of the design

R. Bernero explained that the deterministic or standard review takes account of the Appendix I ALARA concept which might have a reverse ratchet effect for a standard design such as GESSAR II. In other words, a system such as the hydrogen control system on a BWR 6 may be deleted as being unnecessary. He indicated that some USI's can be resolved on a plant-specific basis without much controversy since they are so close to final generic solution.

R. Bernero indicated that the severe accident policy covers both prevention and mitigation. However, the Commission safety goals do not have a specific set of containment performance criteria. Proposed future plants do not span the spectrum of light water reactor designs that exist in the U.S. today. New designs are of only two types - PWRs with large dry containments or BWRs with Mark III containments. These are the most tolerant forms of containments that exist in the spectrum of designs in the U.S. Broader and less tractable uncertainties like the real significance of seismic risk, the real significance of sabotage risk, or the general level of reliance on operators and vulnerability from operator error remain as unanswered questions. D. Okrent noted that in the Sizewell B case, quite a few additional features were added to the plant to reduce the likelihood of core melt. These additional hardware changes were reviewed by the Staff and found not to be cost beneficial. He asked how a modified SNUPPS reactor would fare under a severe accident review. R. Bernero explained how the Sizewell B review was conducted by the Staff. He mentioned the prioritization of a list of modifications where the NRC Staff evaluated and listed them in order of their relative cost benefit. He indicated that it would be his expectation that with changes, a modified SNUPPS plant would be found to have an acceptable design for 10 year certification.

M. Ruben, NRC Reliability and Risk Assessment Branch, described the Staff's use of the GESSAR II PRA in its process of severe accident evaluation (See Appendix XX). He indicated that the PRA was evaluated at Brookhaven National Laboratory. Among the general insights that came out of the PRA study was a revised core melt frequency for internal events of approximately 4×10^{-5} per year for the "average"

site. He noted that approximately 85 percent of the core melt frequency is attributed to a loss of offsite power transient (station blackout). J. C. Ebersole asked if this core melt frequency was without inclusion of the ultimate plant protection system. M. Ruben explained that the ultimate plant protection system (UPPS), a system for providing core makeup water, vessel depressurization, and containment venting on the loss of all AC power, when considered part of the plant, reduced the core melt frequency by a factor of 4 to 5 for internal events.

M. Ruben indicated that the seismic evaluation, carried out separately and submitted slightly later than the internal events PRA, yielded a core melt estimate of approximately 7×10^{-5} per year dominated by postulated seismic relay chatter impact. He explained that while there was some relay fragility data available on chattering, the Staff did not have a detailed study of the plant response to relay chatter. In GE's study some rather simplistic assumptions were made on the failures of safety systems from chattering. A sensitivity study was provided which discussed the impact on various human recovery rates as well as various relay fragility values. He thought that the GE result will probably yield an upward bound on the severity of relay chatter. Elimination of relay chatter impact from the seismic core melt frequency would essentially eliminate that failure mode completely by a factor of two and one-half to three. J. C. Ebersole suggested that the solution seems almost trivial in this case. Make the UPPS seismically qualified and forget about relay chatter. M. Ruben indicated that that is an area under consideration.

R. Villa, General Electric presented a brief description of the GESSAR II nuclear island arrangement (See Appendix XXI). C. Michelson indicated that he agreed with GE's approach to controlling the design of the safety-related areas of the plant. He noted that one of the most important safety-related features involves the coolant water that supplies all of the emergency equipment. He noted that the cooling water to the safety-related equipment was outside GE's scope. R. Villa noted that the cooling water supplies are considerably more complicated to envelope than even the siting envelope. He did point out that GE defines specific requirements for the cooling water that the applicant must satisfy. C. Thomas, NRC Staff indicated that as long as the vendor supplies interface criteria that the utility/applicant has to meet, the Staff has historically been satisfied. J. C. Ebersole asked if there are reliability requirements for the auxiliary feedwater. C. Thomas indicated that he thought that the reliability of the service water system was given in the PRA. J. C. Ebersole asked what it was. D. Hankins, GE, indicated that the reliability criterion for the auxiliary feedwater is a failure rate of 10^{-6}/year. J. C. Ebersole asked if there was a requirement on the continuity of service water flow. R. Villa indicated that GE does have a requirement that the Applicant meet the GE interface requirements.

M. Ruben indicated that the Staff, in its evaluation of the PRA, has identified areas where they are not convinced that interface criteria are complete. He stated that the Staff has required that when a plant

is actually constructed, for those portions with which the utility/Applicant and the architect engineer are involved, the applicant should identify critical components and validate that the reliability of their inplace systems are compatible with those assumed in the GE PRA. D. Okrent suggested that the Staff is not satisfied that the interface requirements for the balance of plant are adequately specified at this time by General Electric. He wondered why the Staff would given final design approval to the GESSAR II design when the requirements is to be placed on the balance of plant are incomplete. M. Ruben indicated that the Staff is relying on GE's reliability criteria as being sufficient until interface criteria for those portions of the plant, not yet designed, are specified.

F. Rowsome, NRC, indicated that Staff in its safety evaluation report intends to offer the licensee an option. He may either verify that his as-built plant is enveloped by the assumption that the Staff is making in its own risk assessment or have the alternative of reopening the risk assessment to attempt to justify other numbers. The Staff is not stipulating reliability criteria or seismic criteria beyond the design basis because the Staff is not striving for a particular safety goal.

J. C. Ebersole asked whether GE would consider an air cooling module to cool the plant. He thought it much more reliable than having to rely on cooling water from a fresh water body such as a creek all of the time. R. Villa expressed concern that GE would have to re-enter the licensing process again if they were to change the configuration of the buildings for air cooling. He stated that air cooling is not part of this design.

R. Villa compared GESSAR II regarding principal design features to other BWR facilities such as Perry, Clinton, and Grand Gulf. P. G. Shewmon noted that the containment leak rate of GESSAR II is five times that of Perry. D. Hankins indicated that GE is offering the applicant a plant which has an allowable one percent leak rate. If the applicant wishes to have a smaller site boundary he must opt for a tighter containment. C. P. Siess questioned whether the utility is buying a poorer containment or a better operating program for meeting leak-rate requirements. R. Villa answered C. P. Siess by indicating that the hydrogen control system has been added as a design requirement for GESSAR. GE has increased the containment strength and provided an allowance for venting of the containment. Also proposed is the UPPS as an alternate to the hydrogen control system. J. C. Ebersole asked why the GESSAR plant has only a 35 percent steam bypass capacity. R. Villa indicated that this was deemed to be a cost-effective number. It is very expensive to have 100 percent bypass. D. Foreman, GE, indicated that a typical reason for picking 100 percent bypass is to increase the capacity factor of the plant. It is strictly economics. With 100 percent bypass capacity, one can more easily bring the plant back to power. D. Okrent pointed out that it has certain safety implications. There is a group of transients that no longer become ATWS candidates.

R. Villa discussed the scope, insights, and results of the GESSAR PRA (see Appendix XXII). He indicated that GE has met the first requirement for the FDA -- compliance with current Commission regulations. This includes TMI requirements as defined in the CP rule, the hydrogen control system, and provisions for venting the containment all of which have been incorporated into the GESSAR design. He indicated that GE has addressed resolution of unresolved and generic safety issues including reliability of decay heat removal systems as well as reliability of AC and DC power supply systems. GE has completed a probabilistic risk assessment considering severe accident vulnerabilities and their added impact on the public health and safety. C. Michelson asked for an explanation of how flooding was evaluated on the intake structure (service water system). K. W. Holtzclaw, GE, indicated that the flood evaluation was done by looking at flood sources on a compartmental basis within the plant. He indicated that GE did not consider the intake structure. C. Michelson asked if GE had considered external floods. K. Holtzclaw indicated that they did not, even though there are some prescriptions in the site envelope requiring the applicant to consider potential external floods. H. W. Lewis asked how uncertainty analysis was included for internal events. D. Hankins indicated that Monte Carlo evaluations were made using the SPASM code.

R. Villa indicated that it was found from a deterministic review that transient initiators were more important for GESSAR in terms of overall risk than LOCA's. The dominant sequence found was station blackout. It was also found that emergency procedure guidelines reduced the likelihood of human failure for the GE design.

R. Villa noted that it was found that the hydrogen control system on GESSAR was judged by GE not to be cost-beneficial by orders of magnitude. D. Hankins explained that hydrogen control had no effect on prevention of core damage or the core damage frequency. Its only effects were in the mitigation area, both in terms of offsite or onsite property damage and offsite risks in terms of man-rem averted. R. Villa stressed that if the mission of the PRA is to reduce the dominant contributors to risk it would be most proper to add the UPPS proposed by GE in lieu of the hydrogen control system. D. Okrent indicated that he found GE's use of the claim of very low core melt frequency as the basis for reducing the containment capability quite troublesome. He suggested that GE is using uncertain aspects of its ability to prevent a core melt to weaken another part of the defense-in-depth. J. C. Ebersole indicated that it was his belief that the hydrogen control igniter system such as that initially developed at TVA was only of nominal benefit at best and ought not to be compared to GE's UPPS which has considerably more potential.

C. Michelson asked how human factors were handled in the PRA other than for sabotage. K. Holtzclaw indicated that GE utilized the Swain-Gutmann guidelines and also tried to reflect as accurately as possible the emergency procedure guidelines that came out of post TMI activities. C. Michelson expressed interest in how GE handled equipment maintenance from the human factors point of view. He

suggested that availability of equipment may be a major contributor to risk.

M. Rubin, NRC, discussed the Brookhaven National Laboratory's (BNL) reassessment of the GE GESSAR PRA. He indicated that there were a number of modifications made in the course of the BNL/Staff review which include such items as additional dependencies between safety functions and some changes to fault trees and event trees. Common mode failure modeling of diesel generators was modified. The reactor protection system mechanical system failure was assessed to be about the same by BNL but the electrical system analysis was modified. Transient frequencies and the loss of feedwater turbine trip frequency was reassessed upward. Some containment event trees were modified and the seismic analysis insight hazard functions, fragilities, and the seismic system analyses were all changed. The results were different numbers found by BNL.

M. Rubin indicated that the dominant sequences were generally the same for GE and the Staff/BNL reassessment. The loss of offsite power sequence dominated. The seismic analysis by BNL yielded a risk factor 4 to 5 times higher than that developed in the GE internal analysis. There were a number of uncertainties regarding both the site hazard function, component fragilities, and the actual selection of components used in their analysis. He indicated that the NRC and BNL considered their fragilities more realistic and believe that GE's conjectures are too optimistic. P. G. Shewmon asked if GE's UPPS was included. M. Rubin indicated that the UPPS impact is not reflected in the numbers from the BNL reassessment. D. Okrent asked if the NRC evaluated the UPPS for internal events. M. Rubin indicated affirmatively and that it made a difference of a factor of 4 to 5.

M. Rubin indicated that that the Staff and BNL found that the events analysis was relatively complete and well done by GE with regard to the present state of the art of PRAs. However, the Staff believes that its reassessment was more reflective of the actual plant. He noted that the seismic evaluation resulted in significant upward reassessment regarding core melt contributions. There were changes in seismic systems analysis. Assessment of relay chatter, fragilities, and site hazards were modified considerably. The seismic core melt contribution, however, was roughly equivalent to that for internal events. H. W. Lewis asked if the Staff modifications were as a result of mistakes or oversights found with regard to GE's evaluation of component failure rates or fragilities. M. Rubin indicated that the majority of cases were differences of opinion on the hypothetical component fragilities. He suggested that the fragilities of components were not evaluated in enough depth by GE and this bears further study. He noted also that the fragilities are site dependent and that fact has to be incorporated in the PRA. This is because of the geological character of the site. Whether the plant sits on soil or bedrock will influence the propagation of acceleration values.

V. AEOD Activities (Open)

[R. F. Fraley was the Designated Federal Official for this portion of the meeting.]

J. Heltemes, Director of AEOD, presented a status of selected reactor program activities in AEOD (See Appendix XXIII). He noted that the LER reporting rule has been in effect since January 1, 1984 and the level of detail in reports is still disappointing. When referring to the nuclear plant reliability data system (NPRDS), he mentioned that NRC contractors do not have access to NPRDS data a fact that often seriously handicaps their project activities for NRC. He noted that in the area of transient patterns program activities, AEOD has evolved a methodology which was published in September 1984. He mentioned three tasks completed in 1984 -- a pilot study analysis of unplanned reactor scrams (the first quarter of 1984), a study of feedwater transients for Westinghouse PWRs covering the period January 1981 to June 1983, and an analysis of 1981 LERs. He explained that since the 1981 LER study did not have sufficient data to allow a trends-and-patterns analysis, the study was expanded to a three year period (1981-1983). He indicated that an analysis of unplanned safety system unavailability for 1984 is one task under development. A proposed analysis of technical specification violations for 1984 will deal with situations that resulted in shutdowns required by technical specifications and operations prohibited by the technical specifications.

H. W. Lewis asked if the analysis of technical specifications violations will be directed toward assessing them in terms of their safety significance. J. Heltemes indicated that that would be one of the focal points of the study. J. C. Ebersole asked what AEOD is doing in the area of generating reliability data on the operation of valves under duress. J. Heltemes pointed to AEOD's active support of a test program on signature analysis of valves. The results of this program are just now becoming available.

J. Heltemes reviewed the status of technical studies as evidenced by documents issued in calendar year 1984. Technical studies were characterized as case studies (a significant safety concern has been identified), engineering evaluations which pertain to safety concerns of less significance, technical reviews of a potential concern, special studies such as reports to the Commission, and program support studies (AEOD's transient patterns programs results). He pointed out that AEOD's resources are apportioned in the following manner: Twenty-Five percent of engineering resources went to case studies, sixty percent went into engineering evaluations, and fifteen percent was used for technical reviews. Special studies resources and program support studies resources are tracked separately since these derive from other parts of the NRC organization.

J. Heltemes noted that the Committee had expressed an interest in indicators of performance which AEOD had developed. He expressed his belief that operational experience should be a valid and direct measure of performance. He presented a list of proposed potential operational experience performance indicators derived from LER data. He pointed out that while he thought counting LERs is a wrong

approach, counting events by looking into the LERs for an in-depth appraisal of performance would be useful. He indicated that AEOD expects to develop indicators of performance routinely for every plant. These include reactor protection system actuations that caused SCRAMs, actuations of other ESF systems, system unavailability and the loss of safety systems, conditions prohibited by technical specifications, and shutdowns required by technical specifications, airborne and liquid releases, and voluntary reports. G. A. Reed noted surprise at the indicators selected since they were not the traditional indicators of good performance such as forced outage rate, capacity factor, and availability factor. He indicated that he was not sure where the tracking of such LER data will correlate with sound performance. J. Heltemes suggested that these indicators from operational experience would be summed with other indicators. They are not meant to be a set by themselves since operational experiences are just one valid indicator of performance.

J. Ebersole noted that J. Mac Evoy, ACRS Fellow, had been independently developing a list of parameters. J. Mac Evoy indicated that he had looked at five parameters. These were plant availability, airborne curies discharged, total man-rem occupational exposure, average yearly plant efficiency, and total number of trips at power reported for the year. He indicated that these were factored into an equation to give plant performance. LERs were deliberately ignored because of the desire to go back into the past as far as possible to compare past performance to present performance and look at indicators that would compare all plants to one another and would not be subjective. G. A. Reed indicated that he was surprised that AEOD had only focused on such a small piece of the puzzle and not concerned itself with the more traditional indicators that have been put forward by INPO as best performance indicators.

C. Michelson noted that the NPRDS was to be a piece of the former LER system that would cover single component failures which would no longer be reported as LERs. They would be reported in the NPRDS data base and would then be available for studies by individuals who were looking at single component failures. He expressed concern about the fact that the information is not at all readily available even to the NRC. While NRC personnel can get the information directly, NRC contractors must go through a second or third party in order to get the information to prepare a report. He also noted that AEOD has been prohibited from publishing certain results derived from data from the NPRDS. He thought this was becoming an unacceptable situation which the full Committee should explore. J. Heltemes indicated that this is not a decision made by AEOD. It was a determination by the NRC that INPO has met all the legal and technical tests for protecting commercial information. AEOD has been forced to implement the consequences of that decision. J. Heltemes made a point of clarification that single failures were never reportable to the NRC under LERs. He explained that many single failures came in under LERs because nuclear plants would enter an LCO which was reportable. C. Michelson pointed out, however, that the number of LERs has been halved since the new LER rule went into effect because the single component events that were once inadvertently in the old LER system

has predominantly gone over to NPRDS. Some of this information has just been lost, but a large fraction of it is now in NPRDS, no longer readily available to the agency. It is not available to the public. Chairman Ward indicated that this may be a topic for future discussion. C. Michelson suggested that this issue be discussed at a meeting with INPO. Perhaps INPO would be willing to provide a list of acceptable contractors who could have access.

VI. Emergency Core Cooling Systems (Open)

[Note: P. A. Boehnert was the Designated Federal Official for this portion of the meeting.]

D. A. Ward indicated that 5 topics were discussed at the ECCS subcommittee meeting on February 21, 1985. Among these were the NRC-RES proposed revision of Appendix K to 10 CFR 50.46, the ECCS Evaluation Model (EM) Requirements, and a recent proposal by Duke Power Company to delete use of the upper head injection (UHI) portion of the ECCS sytem at McGuire (Westinghouse Ice Condenser Plant). Also discussed were the effects of insulation debris generation and possible blockage of containment sumps which could impact pump NPSH margins, a discussion of the status of NRR reactor system branch activities, and a discussion of recommendations and requests on the joint NRC/B&W owners group/EPRI/B&W integral system test program.

D. A. Ward indicated that a half-billion dollar research program has confirmed that the Appendix K basis is conservative. RES has said that the two objectives of the Appendix K rule change are to permit higher power peaking by utilities which will yield a financial savings to the utilities through more efficient use of the core and to generally improve safety through the use of best estimate calculations which will yield a more realistic view of reactor system performance. The approach favored by RES is to allow best estimate calculations to be used in licensing per the SECY-83-472 approach approved by NRR. The old Appendix K will remain (with minor revisions) for use by licensees who find no particular benefit in using the new alternative. He indicated that the Subcommittee was in general agreement with the staff's approach.

Duke Power is proposing to remove UHI from its McGuire plant. D. A. Ward indicated that UHI, a feature of some Westinghouse ice condenser plants, was inserted by the designers to allow greater operating flexibility (higher power peaking factors) without exceeding the 2200°F PCT limit of Appendix K, given a LOCA. He indicated that there are 10 ice condenser plants of which 8 have UHI. The staff expects a request for exemption from the remaining plants.

D. A. Ward mentioned a discussion by B. Sheron of NRR regarding the status of the on-going revision of the upper plenum injection (UPI) ECCS models for the Westinghouse two loop plants. Early ECCS models did not appropriately account for UPI related phenomena. The affected licensees have solicited their respective vendors to develop new EM's that satisfy NRC's concerns.

C. Michelson voiced concerns regarding the effects of fiberglass insulation on containment sump and recirculation sump performance, post LOCA. His concerns are that insulation debris can possibly block the containment sumps and pump/valve seals and impact NPSH margins. Another concern expressed was the use of fiberglass insulation on high energy lines to replace mirror insulation. A LOCA jet can cause severe damage to the fiberglass insulation in the vicinity of the break and the transport of the damaged material is a key question. D. A. Ward indicated that the Staff's presentation was incomplete and the issue will be discussed again by the Subcommittee.

D. A. Ward indicated that the Subcommittee discussed in round table fashion the joint NRC/B&W owners/EPRI/B&W integral system test program and related programs. ACRS consultants such as I. Catton have previously expressed reservations especially regarding the fact that a secondary blowdown accident scenario has not yet been considered. He indicated that I. Catton suggested that combined secondary side/primary system blowdown should be considered in a followup program at the MIST facility.

VII. Radioactive Waste Management and Disposal (Open)

[Note: O. S. Merrill was the Designated Federal Official for this portion of the meeting.]

D. W. Moeller explained that the ACRS subcommittee on waste management met on February 15-16, 1985 to review the following documents.

- Proposed Amendments to Licensing Procedures, 10 CFR 60, Disposal of High-Level Radioactive Waste in Geologic Repositories
- NRC's QA Review Plan for Site Characterization
- Three NRC Generic Technical Positions on High-Level Waste Management: computer codes, waste package performance after repository closure, and post emplacement monitoring

He recommended that Subcommittee consensus reports on these issues be made a part of the full committee minutes (see Appendix XXIV).

D. W. Moeller explained that several members of the Subcommittee misinterpreted the proposed amendments to 10 CFR 60 in assuming that the NRC would not issue a site characterization analysis. He suggested that the Staff ought to rewrite the proposed rule to clarify its role on site screening and characterization and to point out that it is only the draft site characterization analysis that will not be issued. He said that the Staff will issue the analysis in final form. He noted that paragraph 60.17 part (a)(3) leaves open the question as to what restoration is required for a site which, although suitable, was not selected for use as a repository. He indicated that the Subcommittee thought that the phrase "site restoration" should be restored to the draft because it is a more accurate description than "decontamination and decommissioning" which is currently in the document.

In answer to additional views of Commissioner Asselstine on 10 CFR Part 60, D, W. Moeller indicated that the Subcommittee suggested that the NRC monitor DOE's application of the guidelines to see that DOE nominates the best site. The Staff should not independently evaluate the relative merits of the sites. He indicated that the Subcommittee believes that if the EPA standards are met, the NRC should be satified 56with the site nominated by DOE. He referred to a second view of Commissioner Asselstine regarding issuance of site characterization reports in draft form. He indicated that the Subcommittee thought that because of the many existing opportunities for public comment within the review process, issuance of a site characterization analysis in draft form is not necessary. D. Okrent asked if the full Committee has reviewed the EPA's proposed standards for site criteria. D. W. Moeller indicated that the Waste Management Subcommittee has reviewed them as they have developed. H. W. Lewis expressed concern that the EPA standards may be too stringent for NRC to apply.

M. Bell, NRC Deputy Director for Waste Management, expressed concern regarding the ACRS initiating review of the EPA site criteria a year after closure of the public comment period. He did believe that ACRS comments with respect to a discussion including NRC and EPA would be helpful at this stage. D. Okrent noted that EPA is making nontrivial changes in these siting criteria and he thought that the ACRS should review the situation.

D. W. Moeller indicated that M. Bell would like a letter endorsing the current NRC positions with regard to the proposed amendments to 10 CFR Part 60. M. Bell noted that the EPA standards deal only with the cumulative population doses over a long period of time. The EPA standards should treat individual doses especially with regard to drinking water exposure. The issue of protection of the maximum individual still remains untreated by EPA. He noted that the EPA standard was last submitted for public comment in late 1982. D. W. Moeller indicated that a meeting of the Waste Management Subcommittee will be held to review and discuss the EPA standards.

D. W. Moeller introduced a draft letter regarding the ACRS role in the civilian high-level radioactive waste management program. He indicated that the Commission wants the ACRS to sign off on the program. The Committee must decide whether it will handle this matter in the full Committee, the Waste Management Subcommittee, or recommend that the Commission set up a separate advisory committee to advise them on the matter. Also of importance would be the stage at which deliverables would be reviewed. M. Bell indicated that the NRC Staff and DOE are proceeding as if the ACRS will provide advice. D. W. Moeller noted that it is not part of the ACRS charter. D. A. Ward suggested that the Committee position should be developed in a letter. He thought that the Committee ought to ask the Commissioners about the depth or level of detail of review they expect. C. P. Siess expressed concern regarding the full Committee's ability to review DOE and Staff documents in the same depth as the Waste Management Subcommittee. D. Okrent stated his belief that the civilian radwaste issue is one of the more important matters involving the NRC over the next few years. He thought the ACRS ought to attack the issue by hiring more

consultants. P. G. Shewmon indicated that he failed to see that there were many areas that impacted the public health and safety. He thought the program was principally a political issue and deserved a more superficial review. G. A. Reed thought that this project might dilute ACRS priorities. He was in favor of encouraging the Commission to set up a separate advisory committee. D. Okrent was opposed to presenting several options in a Committee letter. D. A. Ward thought the letter should focus on developing a Committee position. No decision was reached on the proposed letter at this meeting. It was decided that the full Committee would reconsider it at the 300th meeting, April 11-13, 1985.

VIII. Evaluation of Nuclear Facility Accidents (Open)

[Note: A. J. Cappucci was the Designated Federal Official for this portion of the meeting.]

H. W. Lewis indicated that a meeting of the Regulatory Policy and Practices Subcommittee was held on March 6, 1984 to review the final report prepared by Brookhaven National Laboratory (BNL) concerning the need for a "NTSB like" organization for the NRC. He noted that the Commission is still reviewing the BNL report and a copy of the report has been sent to the U. S. Congress. He noted that the Subcommittee was unanimous in its recommendation up to a point. He cited statements from the Brookhaven reports which indicated that BNL thought that major NRC accident evaluations are done in a competent manner but ought to be undertaken in a more structured and coordinated approach. Also missing is a separate investigation regarding the determination of the root causes of accidents. He mentioned that the BNL report recommends the setting up of a quasi-independent statutory "Office of Nuclear Safety" which would report directly to the Commission. One of many questions to be settled is how many events during a year are to be considered significant for review by such an Office. It was suggested that, during 1984, there were 8 major occurrences and about 20 events that might qualify as significant.

H. W. Lewis indicated that the Subcommittee was in favor of the Brookhaven approach. Two members thought that the Office of Nuclear Safety (ONS) would not be sufficiently independent of the NRC itself. H. W. Lewis suggested that the ACRS should recommend endorsing the Brookhaven recommendations provided that the ONS operate in an NTSB mode. He thought the NRC ought to take the next step which would involve setting up a planning group outside the NRC.

H. W. Lewis mentioned options involving AEOD, one of which was to absorb the current AEOD responsibilities in the new ONS. He indicated that it was his impression from the report that there was definite pressure to create a new organization with highly dedicated professionals. W. Kato, BNL, indicated that the BNL recommendation did not require statutory authority to set up the ONS unless the NRC wanted it specifically. H. W. Lewis mentioned that there are potential apparent conflicts of interest for C. Michelson and himself which should be factored into the discussion. F. J. Remick thought that, if this independent statutory office had oversight functions, it

must be more independent of the Commission. One would need "go" team investigators. He also thought there ought to be enhancement in the party system with possibly a utility member of a plant safety committee as a designated representative to deal with AEOD. J. C. Ebersole endorsed a group with a distinct oversight function which could review the entire regulatory process. C. Michelson noted that AEOD has done a good job up to the present but he acknowledged that a new group possibly could do better. C. J. Wylie thought that any faults that AEOD has could all be remedied but there would always be questions remaining regarding their independence. G. A. Reed indicated his support of a planning group for such an ONS quite independent of the NRC. D. W. Moeller supported this idea. D. A. Ward indicated that he was not yet convinced that there was a compelling need for a quasi-independent statutory office. H. Etherington suggested that this Office of Nuclear Safety ought to function similar to a volunteer fire department where the highly dedicated professionals would be called when needed.

IX.

DELETION 1

X. Steam Generator Overfill (Open)

[Note: R. Savio was the Designated Federal Official for this portion of the meeting.]

D. Okrent noted that there was interest on the part of some ACRS members in the problem of steam generator overfill transients. J. C. Ebersole indicated that there was a rather high probability of this

event. D. Okrent noted that there is apparent disinterest on the part of utilities. It is a well advertised transient. The question is will main steam lines fail from static loads. He thought that a mandatory response from licensees was in order especially since there has been severe damage from such events in Europe. C. Michelson was skeptical of steam lines falling down from static loads. He was concerned about sagging of pipes and their effect on tearing loose of smaller lines. Also of concern would be the operability of valves after such an event and the effect of overfill on the auxiliary feedwater water turbines. H. Etherington thought it very remote for main steam line pipes to fall under static load since the safety margin is greater than two.

T. Marsh, NRC, presented an overview of the staff program on steam generator overfill transients. He noted that there is no separate program and agreed that overfill can occur either from tube failure (being handled by a list of outstanding questions to owners groups) and control system failure (USI A-17). He agreed that safety relief valve performance ought to be examined and he noted that water hammer events are being treated in USI A-47 under feed control malfunctions. D. Okrent thought that this fragmented approach was not a systemmatic effort. He acknowledged that the issue may be addressed piecemeal under different programs and that issues are addressed as they come up. T. Marsh suggested that there is a program previously stated that is looking at many aspects of this problem of interest to the ACRS but not all aspects. J. C. Ebersole suggested that the utilities know about the issues but are not particularly concerned. T. Marsh mentioned a general information letter that was to be issued by the Staff with a non-mandatory response from applicants. He suggested that such a letter be withheld until the generic implications of the issue are fully known.

N. Anderson, Section Leader in the Generic Issues Branch, briefly discussed the USI A-47, the safety implications of control systems (might lead to an overfill transient). He mentioned work being done at INEL on complete control systems scenarios for Westinghouse and GE nuclear plants. He noted that analytical work on CE and B&W plants is being done at Oak Ridge National Laboratory and Pacific Northwest Laboratories (PNL). PNL is to look at water hammer on a phenomenological basis regarding subcooling. He also mentioned work being done by Creare on water hammer events in main steam lines. D. Okrent requested a copy of the Creare report entitled, "Initial Assessment of Water Hammer in Main Steam Lines During Steam Generator Overfill" (TM1009 dated January 1985). N. Anderson agreed to provide a copy of the report. C. Michelson commented that there are environmental consequences from a rupture of an auxiliary feedwater turbine steam line. D. Okrent requested that members provide comments which could be incorporated in a letter.

XI. Resolution of USIA-44, Station Blackout (Open)

[Note R. Savio was the Designated Federal Official for this portion of the meeting.]

W. Kerr presented an historical prospective on the station blackout unresolved safety issue. He indicated that the last report of the ACRS was in July 1983. He indicated that the NRC Staff was proposing in July 1983 that operating plants be placed into categories of "station blackout vulnerability." The Staff anticipated that the characterization of operating plants would be such that 4 hours, 8 hours and 16 hours of demonstrated station blackout duration capability would be required. W. Kerr indicated that as of February 26, 1985 the Staff now proposes a rulemaking with 4 and 8 hour station blackouts as vulnerability categories. It was indicated that the Subcommittee members thought the Staff proposal reasonable. He noted that a nuclear utility group and owners group presented the opinion that station blackout does not present a level of risk large enough to warrant a rule containing the requirement for the demonstration of a plant's ability to withstand station blackout. They were considering alternative proposals for presentation to the Staff.

T. Speis, NRC suggested a letter from the ACRS endorsing the Staff approach. A. M. Rubin, NRC, indicated that 4 and 8 hours would not be specifically in the rule but in a Regulatory Guide. P. G. Shewmon wondered if the rule would be prescriptive. D. Okrent had specific questions regarding the Staff proposal:

- Were external events included in the frequency of loss of all offsite and onsite power?
- For the 4 hour requirement, what frequency of loss would be considered large enough for plants to have to backfit?
- For the 8 hour category what frequency of loss was considered large enough for plants to have to backfit?
- What is the threshold that requires that plants must take action

D. Okrent asked if use of gas turbines such as those being installed in plants in France would be an alternative solution. A. Rubin indicated that there are other alternatives. The Staff's approach is installation of defense-in-depth. J. C. Ebersole suggested an alternative of an engine-driven charger for the batteries. D. Okrent expressed concern about the seismic vulnerability of such a scheme. D. A. Ward asked how the reactor coolant pump seal failure issue has been factored into the rulemaking. A. Rubin indicated that it is factored into the scenario of longer duration outages.

W. Kerr suggested a possible letter. He pointed out that the ACRS would not be precluded from making further comments since the rule still has to go out for public comment. He thought that the Committee ought to approve the Staff's current approach.

XII. Watts Bar Nuclear Power Plant (Open)

[Note: A. J. Cappucci was the Designated Federal Official for this portion of the meeting.]

J. C. Ebersole indated that TVA will be spending 11 to 20 million dollars to correct problems regarding electrical separation and fire protection requirements at the Watts Bar Nuclear Power plant. He thought this was a very energetic and thorough job which is being driven by Appendix R fire protection regulations and the Branch Technical Position, Appendix A. TVA will be picking up quality assurance problems at the same time. C. Michelson, another Subcommittee member, indicated that there is no undue concern regarding fire protection at Watts Bar. He also noted that Appendix A to the Branch Technical Position is being treated as a rigid rule.

XIII. Systematic Review of Nuclear Power Plants (Open)

[Note: G. R. Quittschreiber was the Designated Federal Official of this portion of the meeting.]

W. Kerr pointed out that on July 18, 1984 the ACRS wrote a letter on the severe accident issue which recommended that a systematic review be undertaken for each nuclear plant now under construction or in operation. D. A. Ward thought this systematic review would not touch on operational considerations or SALP reviews but only review the physical plant. He noted that PRAs have not normally been used in operational scenarios but have been routinely applied to design. There is no comparable method for operational considerations. He suggested and the Committee agreed that a letter be drafted for review at the 300th ACRS Meeting in April.

XV. Executive Sessions (Open)

[Note: R. F. Fraley was the Designated Federal Official for this portion of the meeting.]

A. Subcommittee Assignments

1. Environmental Protection Agency (EPA) Standards for Disposal of High Level Waste

The Committee members discussed what the ACRS role should be regarding the EPA standards for the disposal of high level radioactive waste. It was noted that the ACRS Subcommittee on Waste Management is familiar with these standards and should proceed to review them.

B. ACRS Reports, Letters, and Memoranda

1. ACRS Report on the Nine Mile Point Nuclear Station, Unit 2

The Committee prepared a report to the Commissioners of its review of the application of Niagara Mohawk Power Corporation (the Applicant), acting on behalf of itself and as agent for Rochester Gas and Electric Corporation, Central Hudson Gas and Electric Corporation, New York State Electric and Gas Corporation, and Long Island Lighting Company, for a license to operate the Nine Mile Point Nuclear Station, Unit 2.

2. ACRS Comments on the Proposed Rule on Backfitting

The Committee prepared a report to the Commissioners of its review of Commission progress toward the issuance of a new backfit rule.

3. ACRS Comments on Proposed Amendments to 10 CFR 60, "Disposal of High-Level Radioactive Wastes in Geologic Repositories"

The Committee prepared a report to the Commissioners of its discussion of the proposed amendments to the licensing procedures for 10 CFR Part 60, "Disposal of High-Level Radioactive Wastes in Geologic Repositories."

4. ACRS Comments on the Possibility of an Organization Like the National Transportation Safety Board (NTSB) for Nuclear Safety

The Committee prepared a report to the Commissioners of its deliberations about the need for a formal institutional structure for the investigation of nuclear power plant accidents, patterned after the National Transportation Safety Board (NTSB). The ACRS is actively interested in this project, wishes to be consulted as the issue evolves, and would like to be kept informed. Additional comments by R. C. Axtmann, J. C. Ebersole, M. W. Carbon, W. Kerr, D. A. Ward, and H. W. Lewis were appended.

5. ACRS Comments on the NRC Staff Proposal for the Resolution of USI A-44, "Station Blackout"

The Committee prepared a letter to the EDO regarding its review of the resolution of USI A-44, "Station Blackout," that is being proposed by the NRC Staff. Additional comments by D. A. Okrent were appended.

6. Recommendation for New Generic Issue: Steam Generator and BWR Vessel Overfill

The Committee prepared a memorandum to the EDO regarding its discussion with the NRC Staff on the status of the NRC work on steam generator overfill.

7. Need for Information Concerning Preferred Operator Actions for Postulated Scenarios in Which AC Power is Restored After Severe Core Damage Has Occurred

The Committee prepared a memorandum to the EDO regarding the preferred operator action for postulated accident scenarios such as those where AC power is restored after severe core damage has occurred.

8. ACRS Role in the Civilian Radioactive Waste Management Program

The Committee discussed a proposed report to the Commissioners of its plans for responding to the Commission's need for review of various preliminary matters pertaining to the licensing of the U.S. Department of Energy for the construction and operation of a high-level waste (HLW) repository. Final action on this matter was deferred to the 300th ACRS Meeting (April 11-13, 1985).

C. Generic Issues

1. Systematic Review of Nuclear Power Plants

The Committee discussed proposed clarification of ACRS comments regarding systematic review of nuclear power plants in its report of July 18, 1984 on the proposed NRC statement on severe accident policy. It was agreed to defer final action to the April meeting pending textual contributions from Chairman D. A. Ward and C. Michelson.

D. Future Schedule

1. Future Agenda

The Committee agreed on tentative agenda items for the 300th ACRS meeting, April 11-13, 1985 (see Appendix A).

2. Future Subcommittee Activities

A schedule of future Subcommittee activities was distributed to Members (see Appendix III).

E. St. Lucie Nuclear Power Plant, Unit 2

The NRC Staff has recently informed this office of its plan to approve a maximum power level of 2700 MWt for St. Lucie, Unit 2. The ACRS report on the OL for this plant was based on a power level of 2560 MWt. The power level of Unit 1 was increased to 2700 MWt in 1981. The Reactor Operations Subcommittee Chairman has concluded that no further action regarding this matter by the Committee is warranted. The Committee agreed that no further action was necessary.

F. Waste Management Subcommittee Report

The ACRS Subcommittee on Waste Management met on February 15-16 and reviewed NRC's QA Review Plan for Site Characterization and three NRC Generic Technical Positions on computer codes, waste package performance after repository closure, and post emplacement monitoring. Subcommittee consensus reports on these issues were presented to the full Committee. The Members concluded that the comments of the Subcommittee should be incorporated into the record of the meeting so that they will be available to the NRC Staff.

G. Reactor Protection System Malfunction at Sequoyah Unit 2

During an IE Staff presentation to the ACRS of recent significant events at operating nuclear power plants, I. Villalva, NRC, used a diagram of a Westinghouse UV putput circuit to illustrate a maintenance error which led to the failure of a trip breaker to open automatically upon a valid reactor trip signal. While several questions were directed to the working of that UV output circuit during the session, H. W. Lewis still had unanswered questions after the session and requested that an ACRS staff engineer make an additional inquiry to get answers to specific questions. Dr. Lewis brought several errors associated with the reporting of this incident to the attention of the Committee.

H. Sandia National Laboratory (SNL) Equipment Qualification Work

During the 298th ACRS Meeting, February 7-9, 1985, D. Okrent noted that W. Snyder, SNL Director of Safety Programs, complained during a January 6, 1984 Commission meeting that pressure (indirectly) to control or inhibit reports which affect past or present regulatory decisions has been applied by the NRC Staff at SNL. Since that brief discussion at the 298th Meeting, it has been determined that the matter has been adequately aired by the Commission. The Committee, therefore decided to take no further action.

I. Diablo Canyon Transcripts

The Committee deferred a proposed report to the Commissioners regarding the Commission's denial of the use of safety information on the probability of an earthquake at Diablo Canyon because it did not appear in the hearing record.

J. Meetings with Congressional Representatives

Arrangements and topics for visits with appropriate Congressional representatives have been discussed with the Director, Office of Congressional Affairs, who has recommended that these visits be

deferred until such time as the Committee has substantive comments to present.

K. Rescheduling of December (308th) ACRS Meeting

The December (308th) ACRS meeting was moved forward one week to December 5-7, 1985 at the request of M. W. Carbon.

The 299th ACRS Meeting was adjourned at 2:45 p.m., Saturday, March 9, 1985.

APPENDIX I
LIST OF ATTENDEES

ATTENDEES
299TH ACRS MEETING
MARCH 7-9, 1985

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

David A. Ward, Chairman
Harold W. Lewis, Vice-Chairman
Robert C. Axtmann
Max W. Carbon
Jesse C. Ebersole
William Kerr
Carson Mark
Carlyle Michelson
Dade W. Moeller
David Okrent
Glenn A. Reed
Forrest J. Remick
Paul G. Shewmon
Chester P. Siess
Charles J. Wylie

ACRS Staff

Raymond F. Fraley, Executive Director
M. Norman Schwartz, Technical Secretary
Herman Alderman
Paul A. Boehnert
Anthony J. Cappucci
Robert Cushman
Sam Duraiswamy
Medhat M. El-Zeftawy
John Flack
John T. Gilbert
Elpidio G. Igne
Janet Kotra
Morton W. Libarkin
Richard K. Major
John A. MacEvoy
Thomas G. McCreless
John C. McKinley
Owen S. Merrill
Austin Newsom
Gary R. Quittschreiber
Richard Savio
Stanley Schofer
Shivaji Seth
Alan D. Wang

NRC STAFF ATTENDEES

299TH ACRS MEETING-3/7-9/85

Thursday, March 7, 1985

OFFICE OF NUCLEAR REACTOR REGULATION

R. C. Jones, DSI
R. Hernan, PPAS
E. J. Weinkman, DL
A. Schwencer, DL
T. Novak, DL
D. R. Muller
O. R. Levine
G. Sprkler
G. Thomas
A. Bounia
J. Lazevnick, PSB
R. A. Benedict, DHFS
J. Lane, SDI
G. B. Alderman, DE
D. Beckham, DHFS
R. J. Bosnak

OFFICE OF NUCLEAR REGULATORY RESEARCH

O. E. Bassett
M. Pheishman
P. Ricnap

ANALYSIS AND EVAL. OF OPERATIONAL DATA

J. Heltemes
T. Ippolito
D. Presteman
B. M. Bray

ASLBP

D. L. Preteman

OFFICE OF INSPECTION & ENFORCEMENT

R. F. Heishman

Region I

S. J. Collins

APPLICANT ATTENDEES

299TH ACRS MTG. - 3/7-9/85

Thursday, March 7, 1985

NIAGARA MOHAWK POWER CORP

T. Loomis
D. Barcomb
M. Kammer
E. R. Klein
D. Pike
C. D. Terry
P. Volza
J. Bunyon
S. Wekczek
C. V. Mangau
J. A. Perry
U. J. Donhon
T. E. Lempges
R. P. Burtch
N. Rademacher
B. G. Hooten
C. L. Stuart
F. E. Moyer
R. B. Abbott
A. G. Vierling
M. J. Columb
D. Pracht
D. MacVittie
R. Orr
R. Click
J. Stuttman
A. F. Zallnick
J. M. Maeso
J. J. Bekko
F. J. Giaccio
D. Scobeu

STONE & WEBSTER ENGINEERING CORP.

P. Visalli
C. E. Ader
D. Sutton
B. E. Ebbeson
T. L. Orthier
M. A. Durka
D. A. Boe
M. J. Hazzan
R. H. Pinney
M. Mojibian
M. K. Allen
M. Starchnff
C. Zappile

ROCHESTER GAS & ELECTRIC CORP.

G. Wrobel

CENTRAL HUDSON GAS & ELECTRIC CORP.

C. A. Bolz
S. C. Millspaugh

GENERAL ELECTRIC

P. Yin

PUBLIC ATTENDEES

299TH ACRS MTG. - 3/7-9/85

Thursday, March 7, 1985

H. Renner, NUS Corporation
P. Docherty, Westinghouse
R. Borsum, Babcock & Wilcox
J. Nurmi, EPM
M. Ryan, McGraw Hill
E. Oesterle, Bechtel Power Corporation
T. DiNunno, Self
A. Young
W. Y. Kato
A. J. Pressesky, ANS
J. F. Carew, BNL
E. V. Weinstock, BNL
J. H. Bergla, Pacific Gas & Electric
J. Nurmi, EPM
C. O. Miller, Syst. Safety, Inc.
R. Kubinak, LiLco
S. C. Millspausa, Central Hudson
S. Seth, Mitre
R. Body, KMC
E. Oesterle, Bechtel Power
A. Hengst, Self
K. Troxler, Duquesne Light
F. A. Donath, Earth Technology
E. R. Grant, Gulf States Utilities
J. J. Markham
J. G. McWhorter, Dames & Moore
J. T. Dette, Dames & Moore
C. A. Petrache

NRC ATTENDEES

299TH ACRS MEETING - 3/7-9/85

Friday, March 8, 1985

OFFICE OF NUCLEAR REACTOR REGULATION

L. B. Marsh, DST
N. Anderson, DST
R. Hernan, PPAS
G. Rivenbark, DL
C. D. Sellers, DE
C. P. Tan, DE
G. Lear, DE
D. B. Vassallo, DL
P. T. Kuo, DE
W. Long, DL
J. T. Reard, DL
W. H. Swenson, DL
S. Varga, DL
K. Kniel, DST
A. Rubin, DST
C. Thomas, DL
D. Scaletti, DL
D. Crutchfield, DL
T. J. Kenyin

REGION II

S. Elvad

OFFICE OF INSPECTION & ENFORCEMENT

C. E. Rossi
E. W. Weiss
E. Jordan

APPLICANT ATTENDEES

299TH ACRS MTG. - MARCH 7-9, 1985

Friday, March 8, 1985

GENERAL ELECTRIC CORPORATION

D. A. Hankins
H. Solorzano
D. L. Foreman
K. W. Holtzclaw

PUBLIC ATTENDEES

299TH ACRS MTG. - MARCH 7-9, 1985

Friday, March 8, 1985

J. Nurmi, EPM
P. Docherty, Westinghouse
P. C. Carr, Bechtel Power Corp.
R. Borsum, Babcock & Wilcox
W. Pike, NUS
A. F. Zallwich, Niagara Mohaws Power Corp.
T. Loomis, Niagara Mohawk Power Corp.
L. Connor, NRC Calendar
G. Wrobel, Rochester Gas & Electric
T. Imai, JEPIC
M. Kumada, Kansai
H. M. Fontecilla, Virginia Power
H. M. Fontelilla, Virginia Power
D. A. Repka, BLCP&R
J. D. Trotter, NUS
C. Ader, Stone & Webster Electric Corporation

APPENDIX A
FUTURE AGENDA

APRIL ACRS MEETING

Meeting with NRC Commissioners--Discuss the March 12, 1985 ACRS report on backfitting of nuclear plants, discuss the March 13, 1985 ACRS report on the need for an NTSB to investigate nuclear accidents, and discuss the ACRS role in the civilian radwaste program	2 hrs
Meeting with the American Physics Society (APS) to be briefed regarding their recent report on the "Source Term"	1 hr
Scram Breaker Reliability in PWRs--ACRS comments	3/4 hr
Briefing by representatives of IE regarding vendor inspection program	1 hr
Response to inquiry from Congressman Morris K. Udall dated February 11, 1985 regarding ACRS views, or the views of individual members, regarding the NRC budget request for 1986-87 and any NRC programmatic policies or practices of concern to the ACRS	1 hr
GESSAR II--Continue review of GESSAR II request for a forward-looking FDA	3 hrs
Quantitative Safety Goals--Discuss and comment on NRC Staff's evaluation of the two-year trial period	2 hrs
Systematic Review of Nuclear Power Plants - Discuss clarification of ACRS report of July 18, 1984 regarding the proposed NRC severe accident policy statement	
Long-Range Plan for the NRC--Report of ACRS Subcommittee regarding proposed scope of ACRS study of this matter	3 hrs
ACRS Chairman's report regarding revised ACRS Subcommittee assignments	provide written list of changes
Requalification of Reactor Operators--response to Commissioner Asselstine's request for ACRS comments	deferred
Briefing by Director, Office of International Programs, regarding office activities	deferred
Proposed NRC rule to define, "Items Important to Safety"--ACRS comments	deferred

St. Lucie Nuclear Plant--NRC Staff plans to approve a power level of 2700 MWT for Unit No. 2	no action taken
Amendment to 10 CFR 50, General upgrade of emergency procedures for items such as revised source term	deferred

Reports of ACRS Subcommittees

Subcomittee on Safety Research Program regarding a base NRC Safety Research Program in the future (CPS/SD)	1/2 hr
Subcommittee on Emergency Core Cooling Systems regarding review of resolution of TMI Item II.K.3.30, revision of small break LOCA ECCS Evaluation Models and review of resolution of the RCP trip issues given a small break LOCA (DAW/PAB)	1/2 hr
Subcommittee on Electrical Systems regarding recent NRC actions related to diesel generator reliability (WK/MME-Z)	1/4 hr
Combined Subcommittees on Extreme External Phenomena, Structural Engineering, and Diablo Canyon regarding the status of the NRC Staff's seismic design margin programs and PG&E's plan for a seismic reevaluation of Diablo Canyon (DO/CPS/RPS/EGI/JCM)	1/2 hr
Subcommittee on Procedures and Administration regarding proposed revisions of ACRS Bylaws on activities of members as individuals (JCE/RFF)	

MAY ACRS MEETING

Proposed changes to 10 CFR 50 Appendix K ECCS Evaluation Models

Emergency Planning (consider seismic events)

Emergency Preparedness for Fuel Cycle Facilities

Maintenance Task Action Plan

Revised. 3/9/85

SCHEDULE OF ACRS SUBCOMMITTEE MEETINGS

MARCH

14 (1:00p - 7:00p)	Class 9 Accidents (Wang) - Kerr, Axtmann, Shewmon, Ward. Purpose: To discuss New York Power Authority's Source Term studies.
15	Joint ATWS and Electrical Systems (Boehnert/El-Zeftawy/Savio) - Kerr, Ebersole, Ward, Wylie. Purpose: To review NRC Staff's activities associated with the implementation of the ATWS Rule and the status of NRC actions on scram breaker reliability.
19	Reliability Assurance (Major/Alderman) - Michelson, Ebersole, Reed, Ward, Wylie. Purpose: The Subcommittee will begin its investigation of valve reliability. Programs underway by various NRC offices will be discussed. A presentation by industry on methods to enhance the reliability of motor-operated valves is anticipated. The Subcommittee will also discuss several operational occurrences that involved the failure of valves. The Subcommittee will also allot time to planning its future activities.
20	Electrical Systems (El-Zeftawy/Savio) - Kerr, Ebersole, Lewis, Mark, Michelson, Wylie. Purpose: To discuss recent NRC actions related to diesel generator reliability.
21 & 22	Combined Extreme External Phenomena, ~~Structural Engineering~~, and Diablo Canyon (Culver City, CA) (Igne) - Okrent, Axtmann, Siess, Carbon, Etherington, Lewis, Shewmon, Ward. Purpose: To discuss the status of the NRC Staff's seismic design margin programs and PG&E's plan for a seismic reevaluation of Diablo Canyon.
27	Class 9 Accidents (Wang) - Kerr, Moeller (tent.), Shewmon, Siess (tent.). Purpose: To continue the review of the suite of NRC severe codes. The codes to be reviewed will be the containment codes and MELCOR.
27, 28 & 29	Combined GESSAR II, Reliability and Probabilistic Assessment, and Safeguards and Security (Albuquerque, NM) (Major) - Okrent, Carbon, Ebersole, Etherington, Lewis, Mark, Michelson, Ward, Wylie. Purpose: To continue the Subcommittee's review of GESSAR II for Final Design Approval applicable to future plants. The principal topics to be discussed are plant safeguards and the GESSAR II PRA.

SCHEDULE OF ACRS SUBCOMMITTEE MEETINGS

APRIL

3 — ECCS (Boehnert) - Ward, Ebersole, Etherington, Michelson, Reed. Purpose: To: (1) the W BART/BASH ECCS codes, (2) review the resolution of TMI Item II.k.3.30, revision of small break LOCA ECCS Evaluation Models; and (3) review the resolution of the RCP trip issues given a small break LOCA.

4 — Human Factors (Major/Schiffgens) - Ward, Lewis, Michelson, Moeller, Reed, Remick, Wylie. Purpose: To discuss NUREG/CR-3737, a method of ascertaining management/organization's contribution to safety of operating reactors.

5 — Long Range Plan for the NRC (Major) - Carbon, Lewis, Moeller, Remick, Siess, Wylie. Purpose: To continue discussions on developing a long range plan for the NRC. Also to discuss technical and administrative issues related to the regulation of nuclear power plant safety and safety regulation over the next 5 to 10 years.

9 (Closed) (1:00p) — Procedures & Administration (Fraley) - Ward, Ebersole, Lewis, Shewmon (tent.), Siess. Purpose: To discuss proposed revisions of ACRS Bylaws regarding activities of members as individuals and possible limitations on ACRS activities to accommodate budgetary and manpower resources cuts.

10 (Closed) — Safety Research Program (Duraiswamy) - Siess, Carbon, Kerr, Mark, Michelson, Moeller, Okrent, Shewmon (p.m. only), Remick, Ward. Purpose: To discuss a draft report on the "NRC Safety Research Program" prepared by RES dealing with the justifications for a base NRC Safety Research Program in the future.

10 (12:30p - 1:30p) — State of Nuclear Power Safety (Cappucci) - Kerr, Lewis, Michelson, Okrent, Reed, Shewmon. Purpose: To discuss Subcommittee Charter and proposed schedules.

10 (3:00 p.m.) — Safety Philosophy, Technology, and Criteria (Savio) - Okrent, Ebersole, Kerr, Lewis, Michelson, Remick, Ward, Wylie. Purpose: To review the status of the NRC Staff's evaluation of the trial use of the Commission's proposed Safety Goal Policy.

11 - 13 — 300th ACRS Meeting

17 (Closed) — Joint Reactor Operations and Human Factors (Atlanta, GA) (Major) - Ebersole, Ward, Kerr, Lewis, Michelson, Moeller, Reed, Remick, Wylie. Purpose: To discuss INPO's evaluation of nuclear plant operations and incidents/accidents -- briefing by INPO representatives.

23 — Reliability Assurance (Major) - Michelson, Ebersole, Kerr, Okrent, Reed, Ward, Wylie. Purpose: To continue discussion from the March 19 meeting regarding methods to enhance the reliability of air- and motor-operated valves.

SCHEDULE OF ACRS SUBCOMMITTEE MEETINGS

MAY

2 & 3 — Joint Waste Management and Site Evaluation (Merrill) - Moeller, Axtmann, Carbon, Ebersole, Kerr, Mark, Michelson, Okrent, Shewmon. Purpose: To review the: EPA Standars for HLW Repository; (2) consideration of earthquakes in emergency preparedness; and (3) proposed rule on emergency preparedness for fuel cycle and other radioactive material licensees.

6 (P.M.) — Reactor Operations (Major) - Ebersole, Kerr, Michelson, Moller, Okrent, Reed, Remick, Ward. Purpose: To discuss recent operating occurrences.

7 — Safeguards & Security (Schiffgens) - Mark, Carbon, Ebersole, Michelson, Reed, Siess, Wylie. Purpose: To review the potential consequences of sabotage at nonpower reactors, to be briefed by NMSS on sabotage protection at power reactors, and to hear how the NRC Staff reviews and evaluates licensees' security plans.

8 — Safety Research Program (Duraiswamy) - Siess, Carbon, Kerr, Mark, Michelson, Moeller, Okrent, Remick, Shewmon (p.m. only), Ward. Purpose: To discuss the proposed NRC Safety Research Program and budget for FY 1987 and to gather information for use by the ACRS in its preparation of the annual report to the Commission on the NRC Safety Research Program.

9 - 11 — 301st ACRS Meeting

20 - 25 — Meeting with RSK and GPR (Paris, France & Munich, Germany) (McCreless) - Ward, Kerr, Lewis, Mark, Okrent, Reed. Purpose: To discuss current reactor safety problems and philosophy.

JUNE

4 (tent.) — Regulatory Activities (Duraiswamy) - Siess, Carbon, Kerr, Michelson, Ward, Wylie. Purpose: To review the: (1) proposed General Revisions to Appendix J to 10 CFR 50, "Leak Tests for Primary and Secondary Containments of Light-Water Cooled Nuclear Power Plants," (2) Draft Regulatory Guide on "Containment Leakage Testing," and (3) Reg. Guide 1.23, Rev. 1, "Meteorological Measurement Programs for Nucler Power Plants."

5 — Safety Research Program (Duraiswamy) - Siess, Carbon, Kerr, Mark, Michelson, Moeller, Okrent, Remick, Shewmon, Ward. Purpose: To discuss the updated information (possibly the Budget Review Group mark) on the proposed NRC Safety Research program budget for FY 1987. Also to discuss a draft ACRS report to the Commission on the NRC Safety Research Program and budget for FY 1987.

6 - 8 — 302nd ACRS Meeting

SCHEDULE OF ACRS SUBCOMMITTEE MEETINGS

DATES TO BE DETERMINED	
(April) (tent.)	Qualification Program for Safety-Related Equipment (Cappucci) - Michelson, Ebersole, Reed, Shewmon, Siess, Ward, Wylie. Purpose: To discuss the NRC Staff's resolution of USI A-46, "Seismic Qualification of Equipment in Operating Plants." Also to discuss valve operability with the NRC Staff and industry.
(April) (tent.)	Joint Metal Components and Seismic Design of Piping (Igne) - Shewmon, Axtmann, Etherington, Lewis, Mark, Michelson, Okrent, Siess, Ward. Purpose: To review the NRC Piping Review Committee's overall recommendations on piping system concerns.
(April) (tent.)	Seismic Design of Piping (Igne) - Siess, Etherington, Mark, Okrent, Shewmon. Purpose: To review draft reports issued by the NRC Piping Review Committee on dynamic loads and load combinations and seismic design requirements of piping.
(mid-/late April)	ECCS (Boehnert) - Ward, Ebersole, Etherington, Michelson, Reed. Purpose: To: (1) continue the review of the proposed revision to Appendix K; and (2) continue review of the USI A-43 issue relating to the effects of insulation debris on containment pumps post-LOCA.
(April/May)	ECCS (Palo Alto, CA) (Boehnert) - Ward, Ebersole, Etherington, Michelson, Reed. Purpose: To continue the review of the joint NRC/B&WOG/EPRI/B&W joint IST Program. A visit to the EPRI Stanford Research Institute facilities supporting this Program is also planned.
(May)	Decay Heat Removal Systems (Boehnert) - Ward, Ebersole, Etherington, Michelson, Reed. Purpose: To continue the review of NRR resolution position on USI A-45.
(early June)	River Bend 1 & 2 (location to be determined) (Savio) - Okrent, Ebersole, Shewmon. Purpose: To continue the review of Gulf States Utilities' application for an operating license for the River Bend Nuclear Power Plant Units 1 & 2.
Date To Be Determined	Palo Verde (Maricopa County, AZ) (Wang) - Ebersole, Kerr, Lewis, Wylie. Purpose: To review the final reports for various construction deficiencies and the results of the preoperational testing as requested in ACRS letter dated December 15, 1981.
Date To Be Determined (Closed)	Human Factors Tour (Russellville, AR) (Major) Ward, Lewis, Michelson, Moeller, Reed, Remick, Wylie. Purpose: This will be a tour and examination of ANO-1's emergency procedures (symptom based) and facilities.

SCHEDULE OF ACRS SUBCOMMITTEE MEETINGS

DATES TO BE DETERMINED (CONT'D)

Date To Be Determined (Closed)	Westinghouse Water Reactors (Cappucci) - Ebersole, Etherington, Michelson, Okrent, Siess, Wylie. Purpose: To begin the PDA review of the Westinghouse Advanced PWR (RESAR SP/90).
Date To Be Determined	Joint Reliability & Probabilistic Assessment and Millstone 3 (location to be determined) (Savio/Duraiswamy) - Okrent, Kerr, Ebersole, Lewis, Mark, Michelson, Siess, Ward, Wylie. Purpose: To review the probabilistic risk assessment for Millstone 3.
Date To Be Determined	Fluid Dynamics (Boehnert) - Ward, Ebersole, Etherington, Michelson. Purpose: To review the status of the implementation effort on the hydrodynamics loads issue for GE BWR Mark I-III containments.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
MARCH 14, 1985 (1:00p - 7:00p)	CLASS 9 ACCIDENTS	(WANG) Kerr, Axtmann, Shewmon, Ward

LOCATION: WASHINGTON, DC

BACKGROUND:

Who proposed action: New York Power Authority (NYPA)

Purpose: NYPA proposes to elaborate on their source term studies presented in a previous subcommittee meeting.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. "Source Term Safety Assessment - Indian Point 3," NYPA.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
MARCH 15, 1985	JOINT ATWS and ELECTRICAL SYSTEMS	(BOEHNERT/EL-ZEFTAWY/SAVIO) Kerr, Ebersole, Ward, Wylie Cons.: Davis, Lee, Lipinski

LOCATION: WASHINGTON, DC

BACKGROUND:

Who proposed action: ACRS

Purpose: To review NRC Staff activities associated with implementation of the ATWS Rule and the status of NRC actions on scram breaker reliability.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
MARCH 19, 1985	RELIABILITY ASSURANCE	(MAJOR/ALDERMAN) Michelson, Ebersole, Reed, Ward, Wylie

LOCATION: WASHINGTON, DC

BACKGROUND:

Who proposed action: Michelson

Purpose: The Subcommittee will begin its investigation of valve reliability. Programs underway by various NRC offices will be discussed. A presentation by industry on methods to enhance the reliability of motor-operated valves is anticipated. The Subcommittee will also discuss several operational occurrences that involved the failure of valves. The Subcommittee will also allot time to planning its future activities.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. Memo for J. Ebersole from C. Michelson, Subject: Significant Failure of an RCIC Steam Line Isolation Valve to Open Against Operating Reactor Pressure, 11/14/84
2. Memo for J. Ebersole from R. Fraley, Subject: Significant Failure of an RCIC Steam Line Isolation Valve to Open Against Operating Reactor Pressure, 11/21/84
3. Item Prioritization: Issue 87: Failure of HPCI Steam Line Without Isolation (Draft)

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
MARCH 20, 1985	ELECTRICAL SYSTEMS	(EL-ZEFTAWY/SAVIO) Kerr, Ebersole, Lewis, Mark, Michelson, Wylie

LOCATION: WASHINGTON, DC

BACKGROUND:

Who proposed action:

Purpose: To discuss the status of recent NRC actions related to diesel generator reliability.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
MARCH 21 & 22, 1985	COMBINED EXTREME EXTERNAL PHENOMNA/ [redacted]/DIABLO CANYON	(IGNE) Okrent, Siess, Carbon, Axtmann, Etherington, Lewis. Shewmon, Ward Cons.: Maxwell, Page, Pomeroy, Scavuzzo, G. Thompson, Trifunac, Luco

LOCATION: CULVER CITY, CA

BACKGROUND:

Who proposed action: Okrent/Siess

Purpose: To discuss the status of the NRC Staff seismic design margins programs, and PG&E's plan for a seismic reevaluation of Diablo Canyon.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. PG&E's plan for Long-Term Seismic Program for Diablo Canyon (distributed February 21, 1985).
2. NRC Staff's Seismic Design Margins Program (distributed February 26, 1985).

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
MARCH 27, 1985	CLASS 9 ACCIDENTS	(WANG) Kerr, Shewmon, Moeller (tent.), Siess (tent.)

LOCATION: WASHINGTON, DC

BACKGROUND:

Who proposed action: ACRS

Purpose: To continue the review of the suite of NRC severe accident codes. The codes to be reviewed will be the containment codes and MELCOR.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

To be provided.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
MARCH 27, 28, 29, 1985	COMBINED GESSAR II, RELIABILITY & PROBABILISTIC ASSESSMENT, AND SAFEGUARDS & SECURITY	(MAJOR) Okrent, Ebersole, Etherington, Lewis, Mark, Michelson, Carbon, Ward, Wylie Cons.: A. Camp

LOCATION: SANDIA (ALBUQUERQUE, NM)

BACKGROUND:

Who proposed action: D. Okrent

Purpose: To continue the Subcommittee's review of GESSAR II for Final Design Approval applicable to future plants. The principal topics to be discussed are plant safeguards and the GESSAR II PRA.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. The SER through Supp. 3 is available.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
APRIL 3, 1985	ECCS	(BOEHNERT) Ward, Ebersole, Etherington, Michelson, Reed Cons.: Catton, Schrock, Sullivan, Theofanous, Tien

LOCATION: WASHINGTON, DC

BACKGROUND:

Who proposed action: NRC/ACRS

Purpose: To: (1) review the W BART/BASH ECCS codes; (2) review the resolution of TMI Item II.K.3.30, revision of small break LOCA ECCS Evaluation Models; and (3) review the resolution of the RCP trip issues given a small break LOCA.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

To be provided.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
APRIL 4, 1985	HUMAN FACTORS	(MAJOR/SCHIFFGENS) Ward, Lewis, Michelson, Moeller, Reed, Remick, Wylie
		Cons.: Kris Gimmy (SRL)

LOCATION: WASHINGTON, DC

BACKGROUND:

Who proposed action: ACRS

Purpose: A detailed discussion of NUREG/CR-3737, a method of ascertaining management/organization's contribution to safety of operating reactors. This effort is attempting to identify and collect existing objective measures of plant safety performance and management and organization.

ACRS Fellow, John MacEvoy, has also been attempting to identify safety and performance indicators at operating plants. A discussion of John's work is planned.

A research project on Organizational Analysis and Safety for Utilities with Nuclear Power Plants is tentatively scheduled.

Also planned is a discussion of operator requalification procedures with Headquarters and the Region offices; the topic of "check operators" will be considered.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. NUREG/CR-3737, "An Initial Empirical Analysis of Nuclear Power Plant Organization and Its Effects on Safety Performance." (currently available)
2. Memo for J. Ebersole fm J. MacEvoy, Subject: "Report on Nuclear Power Plant Safety Measurements," dtd November 7, 1984.
3. NUREG/CR-3215, "Organizational Analysis and Safety for Utilities with Nuclear Power Plants," August 1983.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
APRIL 5, 1985	LONG RANGE PLAN FOR THE NRC	(MAJOR) Carbon, Lewis, Moeller, Remick, Siess, Wylie

LOCATION: WASHINGTON, DC

BACKGROUND:

Who proposed action: ACRS

Purpose: To continue discussions on developing a long range plan for the NRC. Discuss technical and administrative issues related to the regulation of nuclear power plant safety and safety regulation over the next 5 to 10 years.

Coordinate discussions for April 11-13, 1985 full Committee meeting. Three hours of discussion time has been requested.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

To be provided.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
APRIL 9, 1985 (1:00 P.M.) (CLOSED)	PROCEDURES & ADMINISTRATION	(FRALEY) Ward, Ebersole, Lewis, Shewmon (tent.), Siess

LOCATION: WASHINGTON, DC

BACKGROUND:

Who proposed action:

Purpose: To discuss proposed revisions of ACRS Bylaws regarding activities of members as individuals and possible limitations on ACRS activities to accommodate budgetary and manpower resources cuts.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
APRIL 10, 1985 (CLOSED)	SAFETY RESEARCH PROGRAM	(DURAISWAMY) Siess, Carbon, Kerr, Mark, Moeller, Michelson, Remick, Ward, Okrent, Shewmon (p.m. only)

LOCATION: WASHINGTON, DC

BACKGROUND:

Who proposed action: C. SIESS

Purpose: To discuss the "NRC Safety Research Program" Report prepared by RES which provides justification for a base NRC Safety Research in the future.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. Draft 1 of the NRC Safety Research Program report, dated November 21, 1984 (was sent to all members on 11/28/84)
2. Draft 2 of the NRC Safety Research Program report (will be sent to members as soon as possible)

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
APRIL 10, 1985 (12:30 pm)	STATE OF NUCLEAR POWER SAFETY	(CAPPUCCI) Kerr, Lewis, Michelson, Okrent, Reed, Shewmon

LOCATION: WASHINGTON, DC

BACKGROUND:

Who proposed action: W. KERR

Purpose: To discuss Subcommittee Charter and proposed schedules.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
APRIL 10, 1985 (3:00 p.m.)	SAFETY PHILOSOPHY, TECHNOLOGY, AND CRITERIA	(SAVIO) Okrent, Ebersole, Kerr, Lewis, Michelson, Remick, Ward, Wylie

LOCATION: WASHINGTON, DC

BACKGROUND:

Who proposed action: NRC Staff/Subcommittee

Purpose: To review the status of the NRC Staff's evaluation of the trial use of the Commission's proposed Safety Goal Policy.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. NRC Task Force report on the two-year evaluation of the trial use of the Commission's proposed Safety Goal Policy (expected in early February).
2. Summary of tentative NRC Task Force recommendations on revisions to and future use of the NRC Safety Goal Policy.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
APRIL 17, 1985 (CLOSED)	JOINT REACTOR OPERATIONS AND HUMAN FACTORS	(MAJOR) Ebersole, Ward, Kerr, Lewis, Michelson, Moeller, Reed, Remick, Wylie Cons.: Kris Gimmy (SRL)

LOCATION: INPO Offices, 1100 Circle 75 Pkwy, Suite 1500, Atlanta, GA
(404) 953-3600
INPO Contact: Terry Sullivan

BACKGROUND:

Who proposed action: ACRS/INPO

Purpose: To discuss INPO evaluation of nuclear plant operations and incidents/accidents -- briefing by INPO representatives. This briefing was endorsed by the ACRS.

This meeting is expected to be closed due to the privileged and proprietary nature of the information to be discussed.

Other Topics as Desired

- assistance activities
- training and accreditation
- evaluations (simulator, SRO Peer evaluations, corporate, etc.)
- INPO relationship with NRC-NUMARC
- NPRDS
- Human performance evaluation
- INPO international participation program

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

Available background will be sent in a status report.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
APRIL 23, 1985	RELIABILITY ASSURANCE (VALVES)	(MAJOR) Michelson, Ebersole, Kerr, Okrent, Reed, Ward, Wylie Cons.: H. Jones (TVA)

LOCATION: WASHINGTON, DC

BACKGROUND:

Who proposed action: ACRS

Purpose: To continue discussion from the March 19, 1985 meeting. Topics of discussion will focus on methods to enhance the reliability of air- and motor-operated valves. Staff, vendor, and utility participation is anticipated.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. Status Report will be provided prior to meeting.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
MAY 2 & 3, 1985	JOINT WASTE MANAGEMENT AND SITE EVALUATION	(MERRILL) Moeller, Axtmann, Carbon, Ebersole, Kerr, Mark, Michelson, Okrent, Shewmon Cons.: Krauskopf, F. Parker, Steindler Others (to be determined)

LOCATION: WASHINGTON, DC

BACKGROUND:

Who proposed action: D. Moeller

Purpose: To review: (1) EPA Standards for HLW Repository,
(2) Consideration of Earthquakes in Emergency Preparedness; and
(3) Proposed Rule on Emergency Preparedness for Fuel Cycle and Other Radioactive Material Licensees.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. U.S. Environmental Protection Agency, Proposed Rule, 40 CFR Part 191, "Environmental Standards for the Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes," Federal Register (47FR58196), Vol. 47, No. 250, December 29, 1982, pp. 58196-58206.
2. U.S. Nuclear Regulatory Commission, 10 CFR Part 50, "Emergency Planning and Preparedness for Production and Utilization Facilities," Federal Register (49FR49640), Vol. 49, No. 247, December 21, 1984, pp. 49640-49643.
3. U.S. Nuclear Regulatory Commission, Proposed Rule, 10 CFR Parts 30, 40, and 70, "Emergency Preparedness for Fuel Cycle and Other Radioactive Material Licensees," (expected to be received by ACRS by mid- to late March).

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
MAY 6, 1985 (P.M.)	REACTOR OPERATIONS	(MAJOR) Ebersole, Kerr, Michelson, Moeller, Okrent, Reed, Remick, Ward

LOCATION: WASHINGTON, DC

BACKGROUND:

Who proposed action: STAFF/ACRS

Purpose: To discuss recent operating occurrences.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. Status report to be provided prior to meeting.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
MAY 7, 1985	SAFEGUARDS & SECURITY	(SCHIFFGENS) Mark, Carbon, Ebersole, Michelson, Reed, Siess, Wylie

LOCATION: WASHINGTON, DC

BACKGROUND:

Who proposed action: Okrent, Michelson, Mark

Purpose: To review the potential consequences of sabotage at non-power reactors and to be briefed by NMSS on sabotage protection at power reactors.

NRC definitions of "sabotage," "malicious mischief," and "vandalism."

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

None

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
MAY 8, 1985	SAFETY RESEARCH PROGRAM	(DURAISWAMY) Siess, Carbon, Kerr, Mark, Michelson, Moeller, Okrent, Remick, Shewmon(p.m. only) Ward

LOCATION: WASHINGTON, DC

BACKGROUND:

Who proposed action: Routine process

Purpose: To discuss the proposed NRC Safety Research Program and budget for FY 1987 and gather information for use by the ACRS in its preparation of the annual report to the Commission on the NRC Safety Research Program and budget.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. Proposed NRC Safety Research and budget for FY 1987 (expected to be made available to the ACRS during April 1985).

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
MAY 20-25, 1985	DELEGATION TO MEET WITH GPR AND RSK	(McCRELESS) Ward, Kerr, Lewis, Mark, Okrent, Reed

LOCATION: Paris, France
Munich, Germany

BACKGROUND:

Who proposed action: Invitation from Group Permanent and Reaktor Safety Commission

Purpose: To discuss current reactor safety problems and philosophy.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. Memo to ACRS from T. G. McCreless dated March 5, 1985 on current status of meetings.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
JUNE 4, 1985 (tentative)	REGULATORY ACTIVITIES	(DURAISWAMY) Siess, Carbon, Kerr, Michelson, Ward, Wylie

LOCATION: WASHINGTON, DC

BACKGROUND:

Who proposed action:

Purpose: To review the following:

1) Proposed General revisions to Appendix J to 10 CFR 50, "Leak Tests for Primary and Secondary Containments of Light-Water Cooled Nuclear Power Plants," and

2) Draft Regulatory Guide on "Containment Leakage Testing."

3) Regulatory Guide 1.23, Rev. 1, "Metèorological Measurement Programs for Nuclear Power Plants."

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

The following are expected to be made available to the ACRS during May 1985:

1. Proposed revisions to Appendix J to 10 CFR 50
2. Draft Regulatory Guide on "Containment Leakage Testing."
3. Regulatory Guide 1.23, Rev. 1, "Meteorological Measurement Programs for Nuclear Power Plants."

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
JUNE 5, 1985	SAFETY RESEARCH PROGRAM	(DURAISWAMY) Siess, Carbon, Kerr, Mark, Michelson, Moeller, Okrent, Remick, Shewmon, Ward

LOCATION: WASHINGTON, DC

BACKGROUND:

Who proposed action: Routine process

Purpose: To discuss the updated information (possibly the Budget Review Group mark) on the proposed NRC Safety Research program budget for FY 1987. Also to discuss a draft ACRS report to the Commission on the NRC Safety Research Program and budget for FY 1987.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. Updated information on the NRC Safety Research Program and Budget for FY 1987 is expected to be made available to the ACRS during May 1985.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
TO BE DETERMINED (APRIL) (tentative)	QUALIFICATION PROGRAM FOR SAFETY-RELATED EQUIPMENT	(CAPPUCCI) Michelson, Ebersole, Reed, Shewmon, Siess, Ward, Wylie Cons.: to be determined

LOCATION: WASHINGTON, DC

BACKGROUND:

Who proposed action: NRC Staff

Purpose: To discuss the NRC Staff resolution of USI A-46, "Seismic Qualification of Equipment in Operating Plants." Also to discuss valve operability with the NRC Staff and industry.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

To be supplied.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
TO BF DETERMINED (April) (tentative)	JOINT METAL COMPONENTS AND SEISMIC DESIGN OF PIPING	(IGNE) Shewmon, Axtmann, Etherington, Lewis, Michelson, Okrent, Ward, Siess, Mark Cons.: Bender, Pickel, Rodabaugh, B. Thompson, Dillon, Kassner, Bush

LOCATION: WASHINGTON, DC

BACKGROUND:

Who proposed action: NRC Staff

Purpose: To review the NRC Piping Review Committee's overall recommendations on piping system concerns.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. Report of the NRC Piping Review Committee on recommendations on piping, NUREG-1061, Vo. V, (available in early April 1985).
2. NUREG-0313, Rev. 2, implementation report on pipe crack (available late March).

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
TO BE DETERMINED (April) (tentative)	SEISMIC DESIGN OF PIPING	(IGNE) Siess, Etherington, Mark, Okrent, Shewmon Cons.: Bender, Pickel, Bush

LOCATION: WASHINGTON, DC

BACKGROUND:

Who proposed action: NRC/Siess

Purpose: To review draft reports issued by the NRC Piping Review Committee on dynamic loads and load combinations and seismic design requirements of piping.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. NUREG-1061, Vol. IV, "Evaluation of Other Dynamic Loads and Load Combination," (report published and distributed).
2. NUREG-1061, Vol II, "Review of Seismic Design Requirements for Nuclear Power Plant Piping," (should be available by March).

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
TO BE DETERMINED (mid- to late April)	ECCS	(BOEHNERT) Ward, Ebersole, Etherington, Michelson, Reed Cons.: Catton, Schrock, Sullivan, Theofanous, Tien

LOCATION: WASHINGTON, DC

BACKGROUND:

Who proposed action: NRC/ACRS

Purpose: To: (1) continue the review of proposed revision to Appendix K, and (2) continue the review of the USI A-43 issue relating to the effects of insulation debris on containment pumps post-LOCA.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

To be provided.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
TO BE DETERMINED (MAY)	ECCS	(BOEHNERT) Ward, Ebersole, Etherington, Michelson, Reed Cons.: Catton, Sullivan, Schrock, Theofanous, Tien

LOCATION: PALO ALTO, CA

BACKGROUND:

Who proposed action: EPRI/NRC

Purpose: To continue the review of the joint NRC/B&WOG/EPRI/B&W joint IST Program. A visit is planned to the EPRI Stanford Research Institute facilities supporting this Program.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

To be provided.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
TO BF DETERMINED (MAY)	DECAY HEAT REMOVAL SYSTEMS	(BOEHNERT) Ward, Ebersole, Etherington, Michelson, Reed Cons.: Catton, Davis

LOCATION: WASHINGTON, DC

BACKGROUND:

Who proposed action: NRR

Purpose: To continue the review of NRR resolution position on USI A-45.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

To be provided.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
TO BE DETERMINED (early June)	RIVER BEND	(SAVIO) Okrent, Ebersole, Shewmon

LOCATION: (to be determined)

BACKGROUND:

Who proposed action: D. Okrent

Purpose: To continue the review of Gulf States Utilities' application for an operating license.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. NRC Staff to supply SER Supplement 2 in early March to support the April 10, 1985 Subcommittee meeting and action at the April 11-13, 1985 full Committee meeting.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
TO BE DETERMINED	PALO VERDE	(WANG) Ebersole, Kerr, Lewis, Wylie

LOCATION: Maricopa, AZ

BACKGROUND:

Who proposed action: NRC Staff/ACRS

Purpose: To review the final reports for various construction deficiencies and the results of the preoperational testing as requested in ACRS letter dated December 15, 1981.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

To be supplied.

SCHEDULE OF ACRS SUBCOMMITTEE TOUR

DATE	SUBCOMMITTEE TOUR	STAFF ENGR. & MEMBERS
TO BE DETERMINED	HUMAN FACTORS (CLOSED)	(MAJOR) Ward, Lewis, Michelson, Moeller, Reed, Remick, Wylie

LOCATION: ANO-1, Russellville, AR (~50 miles outside of Little Rock, AR)

BACKGROUND:

Who proposed action: Human Factors Subcommittee

Purpose: This will be a tour and examination of ANO-1's emergency procedures (symptom based) and facilities. The Subcommittee wants the opportunity to examine procedures at an operating plant and see how the TMI required backfits such as SPDS interface. Up to a day and a half is expected. ANO-1 is an 850 MWe, B&W PWR.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. One copy of Arkansas Nuclear One, Unit 1 Emergency Operating Procedures is available for your inspection at the ACRS Office (ask Mr. R. Major for them).

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
TO BE DETERMINED	WESTINGHOUSE WATER REACTORS (CLOSED)	(CAPPUCCI) Ebersole, Etherington, Michelson, Okrent, Siess, Wylie

LOCATION: Washington, DC

BACKGROUND:

Who proposed action:

Purpose: To begin the PDA review of the Westinghouse Advanced PWR (RESAR SP/90).

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

(To be provided later.)

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
TO BE DETERMINED	JOINT RELIABILITY PROBABILISTIC ASSESSMENT AND MILLSTONE 3	(SAVIO/DURAISWAMY) Okrent, Kerr, Ebersole, Lewis, Mark, Michelson, Siess, Ward, Wylie

LOCATION: (to be determined)

BACKGROUND:

Who proposed action: Okrent/Kerr

Purpose: To review the Millstone 3 PRA.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. NRC report documenting the results of the NRC/LLNL review of the Millstone 3 PRA.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE MEETING	STAFF ENGR. & MEMBERS
TO BE DETERMINED	FLUID DYNAMICS	(BOEHNERT) Ward, Ebersole, Etherington, Michelson Cons.: Bush, Catton, Schrock, Theofanous

LOCATION: WASHINGTON, DC

BACKGROUND:

Who proposed action: ACRS

Purpose: To review the status of the implementation effort on the hydrodynamics loads issue for GE BWR Mark I-III containments.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

NRR STAFF PRESENTATION TO THE ACRS

SUBJECT: NIAGARA MOHAWK POWER CORPORATION
OPERATING LICENSE APPLICATION
FOR NINE MILE POINT NUCLEAR STATION - UNIT 2

DATE: MARCH 7, 1985

PRESENTER: EDWARD J. WEINKAM, III

PRESENTER'S TITLE/BRANCH/DIV: LICENSING PROJECT MANAGER
LICENSING BRANCH NO. 2
DIVISION OF LICENSING

PRESENTER'S NRC TEL. NO.: 301-492-9788

SUBCOMMITTEE: DR. C. SIESS
MR. J. EBBERSOLE

STATUS OF REVIEW

- SER ISSUED FEBRUARY 1985

- FES SCHEDULED APRIL 1985

- SSEP #1 SCHEDULED MAY 1985

- NO HEARINGS SCHEDULED

- CONSTRUCTION COMPLETION SCHEDULED FOR 2/86

OUTSTANDING ISSUES

1) SNOW LOADS
2) RWCU LINE BREAK - CONFIRMATORY
3) PRESERVICE INSPECTION/INSERVICE INSPECTION
4) EQUIPMENT QUALIFICATION
5) STEAM BYPASS - CONFIRMATORY
6) SECONDARY CONTAINMENT BYPASS
7) CONTAINMENT ISOLATION
8) CONTAINMENT LEAK TESTING
9) CONTAINMENT FRACTURE TOUGHNESS
10) POST-ACCIDENT MONITORING INSTRUMENTATION
11) SEPARATION CRITERIA
12) SAFE/ALTERNATE SHUTDOWN
13) ESSENTIAL LIGHTING
14) AIR START SYSTEM
15) OPERATIONS MANAGEMENT
16) PGP's
17) PREOP AND S/U TEST ABSTRACTS
18) DCRDR AND SPDS

OUTSTANDING ISSUE NO. 1
SNOW LOADS

APPLICANT

- ORIGINAL DOCUMENTATION OF DESIGN IN FSAR

  °WEIGHT ON GROUND, 100 YEARS RECURRENCE - 85 PSF

  °48-HOUR PMWP-56 PSF

  °YIELDS EXTREME SNOW LOAD-141 PSF

- REVISED DESIGN BASIS DOCUMENTATION (APRIL 1984 FSAR REV)

  °100 YEAR WEIGHT ON GROUND - 45 PSF

  °NO EXTREME SNOW LOAD CONSIDERATION

STAFF REQUEST

- JUSTIFY REDUCTION FROM 85 PSF TO 45 PSF IN LIGHT OF CURRENT AVAILABLE MET DATA

- PROVIDE EXTREME ENVIRONMENTAL LOAD COMBINATIONS

- CAN CATEGORY I STRUCTURES WITHSTAND ABOVE LOADINGS

OUTSTANDING ISSUE NO. 11

SEPARATION CRITERIA

ISSUE

- CLARIFY WHEN AND WHERE ANALYSIS IS USED TO JUSTIFY LESS THAN REQUIRED 6-INCH SEPARATION IN PGCC CABINETS

OUTSTANDING ISSUE NO. 12

SAFE AND ALTERNATE SHUTDOWN

ISSUE

- STAFF REVIEWING ADDITIONAL INFORMATION ON SAFE AND ALTERNATE SHUTDOWN CAPABILITIES

OUTSTANDING ISSUE NO. 13

ESSENTIAL LIGHTING

ISSUE

- DEMONSTRATE ABILITY TO MAINTAIN LIGHTING IN PLANT SAFETY-RELATED AREAS OTHER THAN THOSE COVERED BY EMERGENCY LIGHTING

OUTSTANDING ISSUE NO. 14

AIR START SYSTEM

ISSUE

- LACK OF DRYERS IN AIR START SYSTEM

NUREG/CR-0660 IDENTIFIES MOISTURE IN AIR START SYSTEM

AS LEADING CAUSE OF DG UNRELIABILITY

NRC REGION I

EVALUATION OF CONSTRUCTION QUALITY

at

NINE MILE POINT UNIT 2

AS OF JANUARY 1985

Presented to ACRS Subcommittee

Syracuse, New York

February 20-21, 1985

Approved:

2-19-85

TABLE OF CONTENTS

NINE MILE POINT 2 CONSTRUCTION

1.0 Introduction

Nine Mile Point, Unit 2, is currently under construction on the Southeast shore of Lake Ontario located northeast of Oswego, New York. The plant is jointly owned by: Niagara Mohawk Power Corp. (41%); New York State Electric and Gas Corp. (18%); Long Island Lighting Co. (18%) Rochester Gas and Electric Corp. (14%); and Central Hudson Gas and Electric Corp. (9%). The construction permit for Nine Mile Point 2 was issued on June 24, 1974. Stone and Webster Engineering Corp. serves as the Architect/Engineer and Construction Manager.

The Nine Mile Point, Unit 2, reactor is a General Electric BWR5 with a net electrical output of 1080 MWe. The primary containment is a Mark II over-and-under concept with a reinforced concrete steel lined pressure suppression structure.

NRC Region I (previously the AEC) began performing inspections at Nine Mile Point 2 in 1972 and has completed about 120 inspections since that time. These inspections involved the observation of work in progress, examination of completed work, examination of work control documents, independent measurements, and the examination of quality records.

This report describes the process used by the licensee to monitor and control construction quality, discusses the results of independent evaluations of the licensee performance, and addresses both the NRC inspection program and the Systematic Assessment of Licensee Performance (SALP) efforts conducted by Region I. Numerous deficiencies have been identified with the implementation of site Quality Assurance (QA) programs and the quality of installed hardware. The Regional staff has confidence that current construction activities are generally in compliance with regulatory requirements. Pending Regional verification of licensee corrective actions in response to both team and routine inspection findings, and closure of existing open concerns, the overall plant construction and compliance to NRC regulatory requirements will be assessed at a future date.

2.0 Independent Evaluations

Aside from those quality assurance/quality control (QA/QC) and project management evaluations that normally take place during major facility construction programs, the applicant has participated in evaluation programs sponsored by independent organizations. The independent evaluations include those sponsored by the Institute of Nuclear Power Operations (INPO) and the Joint Utility Assessment Team (JUAT).

Region I is cognizant of the findings generated during the independent evaluations.

2.1 INPO Evaluations

The licensee and other co-tenants conducted a self-evaluation based upon INPO performance objectives and criteria. The self-evaluation was conducted in September 1982 and approximately 2600 man-hours were expended performing direct process observation, personnel interviews and document review. Problems were identified with QA/QC staffing levels, some areas of design control, segregation of nonconforming material. No substantial hardware deficiencies were identified.

An INPO construction audit was conducted from September 24 - October 5 and October 15 to October 19, 1984. The audit was scheduled to address organizational structure, design control, material storage, RHR system design verification, QA program implementation, equipment qualification, hardware installation inspections, plant safety, and test activities. The formal INPO audit report has not yet been released. Three Construction Deficiency Reports were prepared as a result of INPO concerns, regarding control of equipment spare parts; diesel generator voltage profile study for 600 V Class IE starting loads; and the Auxiliary system voltage profile did not include 208 /120 VAC systems.

The audit identified strong points regarding licensee assessment of contractor's performance and the scope of Stone and Webster Engineering Corporation (SWEC) Engineering Assurance audits. The final report is scheduled to be reviewed by Region I when it becomes available.

2.2 Joint Utility Assessments

The Joint Utility Assessment Team (JUAT) program provides independent audits, by utility senior management, of the licensee QA activities. This form of audit satisfies the requirements of 10 CFR 50 Appendix B Criteria II for regular utility oversight of the status and adequacy of the implementation of the QA program. The audits evaluate the licensee performance, and recommendations are made for program improvement as necessary. To date, there have been four audits of the licensee corporate and site QA efforts. The most recent assessment covered the period from March 5 to 9 of 1984. The scope of the audit included QA coverage of Start-up and Test activities; effectivness of nonconformance trending programs; and timeliness of corrective actions to the NRC CAT inspection. Several enhancements were recommended for improving the trending efforts and resultant application of corrective actions. The audit identified inadequate followup to QA nonconformances in that root causes are not identified and corrective action to preclude recurrence are not specified in the nonconformance report disposition.

3.0 Region I and I&E Headquarters Inspection Program

Region I inspections of construction activities at Nine Mile Point 2 have been conducted in accordance with the program established by the Office of Inspection and Enforcement (I&E). The objective of these safety inspections is to obtain sufficient information through direct observation in the field, personnel interviews, and review of procedures and records to determine

whether construction and installation of safety-related components, structures, and systems meet applicable requirements. A portion of the inspection effort is directed toward inspection of the applicant's Quality Assurance Program and its implementation. The Nine Mile Point Unit 2 QA program for the design and construction phase is described in the Preliminary Safety Analysis Report.

The NRC inspection program is currently performed by both resident and region-based generalist and specialist inspectors. This program has been developed over a period of years to place emphasis on potentially generic deficiencies and on areas experience has shown to be problem areas.

The Nine Mile Point 2 site was staffed with a construction resident inspector from October 1981 to May 1983 and from July 1983 to the present. A second construction resident was assigned in October 1984. In November 1984, an additional Senior Resident Inspector was dedicated on a part-time basis to follow the pre-operational testing program. The direct observation, independent verification, and daily presence of resident inspectors at the facility provide a means to detect quality problems and to monitor the licensee compliance to the site QA programs.

3.1 Inspection History

Initial inspection of the applicant's QA program was performed in 1972. Followup inspections were subsequently performed to verify the implementation of an acceptable QA program. Regional inspection has been performed in concert with the licensee completion of construction activities.

Region I inspections monitored activities including soils and foundations, concrete work, safety-related structures, piping, welding, electrical activities, safety-related mechanical components, instrumentation, and related areas. The enclosure identifies the inspections performed, the areas inspected, and significant inspection findings. At present, about 120 inspection reports have been issued or are pending for the Nine Mile Point 2 facility. A comparison of inspection hours expended at BWR facilities at a similar stage of construction is shown below:

Nine Mile Point 2	Hope Creek	Shoreham	Susquehanna 1	Limerick
8250 hrs.	7600 hrs.	6500 hrs.	7100 hrs.	7800 hrs.

3.2 Enforcement History

The inspection program uses enforcement measures to promote adherence to regulatory requirements, reduce repeated nonconformances, and encourage self-identification and correction of nonconformances. Notices of Violations, have been issued when necessary. The applicant has been required to respond to these Notices of Violation and provide the proposed actions to correct the nonconforming conditions and to prevent recurrence of similar violations. NRC inspectors and management have reviewed and evaluated these responses for acceptability.

The inspection staff confirms, during subsequent inspections, that corrective actions are properly implemented. The following table gives a comparison of the Nine Mile Point 2 enforcement statistics with those of three other plants. Early enforcement actions were classified as "violations", "infractions", and "deficiencies" (in descending order of severity) while the more recent reports contain violations categorized into severity levels ranging from I to V (again, in descending order) and deviations.

Facility	CPPR	VIOL	INF	DEF	I	II	III	IV	V	DEV	TOTAL
Shoreham	4/14/73	0	38	6	0	0	0	17	13	1	75
Susq. 1	11/2/73	0	47	15	0	0	0	18	19	3	102
Hope Crk.	11/2/74	0	19	5	0	0	0	19	13	2	58
NMP-2	6/24/74	0	12	1	0	1	1	29	20	1	64

The early enforcement history did not identify any significant programmatic weakness within the application of the QA program at Nine Mile Point 2. A greater number of recent violations with higher severity levels has been assessed against Nine Mile Point 2. The level II violation was issued for a multitude of QA problems identified during an I&E CAT inspection. A $100,000 Civil Penalty and Order were simultaneously issued to the licensee. The Level III violation was written for using trainees to conduct inspection efforts, and the identification of falsified inspection records. A $100,000 Civil Penalty was simultaneously issued.

Significant deficiencies have been identified with the application of QC inspection programs and the adequacy of nondestructive examination tests. The licensee has instituted numerous corrective actions including program revisions and hardware reinspections. As a result of the I&E CAT Order, a third party has evaluated the adequacy of the implemented corrective actions. The licensee is formulating plans to address the recommendations of that third party audit. The majority of the CAT concerns are currently carried on the open item list, pending Region I closeout. Until the verification closeout cycle is complete, definitive statements regarding the corrective action implementation, in response to numerous deficiencies, cannot be made.

3.3 Regional Construction Team Inspection (RCTI)

For a more in-depth assessment of construction quality, two Regional Construction Team Inspection (RCTIs) have been conducted.

The first inspection was conducted from November 30 to December 18, 1981 by three region-based inspectors, the resident inspector and a section chief. The inspection covered site quality assurance activities; design controls; project management practices; procurement control; and construction controls in the electrical, mechanical and nondestructive examination areas. The inspection function was to assess the licensee's management control of the Nine Mile Point 2 construction activities. The inspection involved 394 on-site inspection hours.

Several problems were identified as noted below.

- Structural steel items improperly released from storage
- Inadequate training of contractor personnel
- Overdependence on contractor construction personnel to monitor quality related activities
- Nonconformances not evaluated to determine root cause
- Untimely corrective action for QA identified deficiencies
- Project Manual does not define position descriptions, leading to confusion on roles and responsibilities
- Licensee has set inequitable pay and benefits for QA personnel and exercised ineffective management control over SWEC and the sub-contractors.
- Inadequate design control measures to assure translation of regulatory requirements into design criteria.
- Cable tray procurement documents did not conform to NEMA standard.

The Notice of Violation associated with the team's inspection report (50-410/81-13) and the SALP Cycle 2 report, transmitted the fact that the observed deficiencies, when viewed in total, are indicative that an inappropriate and ineffective Quality Assurance program had been implemented at Nine Mile Point 2.

Satisfactory observations were identified in the areas of document control; records control; audit programs; design change control; design interfaces; 10 CFR 50.55(e) reporting; receiving inspection; warehouse facility; weld fitups and welding material control.

The second inspection was conducted from December 3 to December 14, 1984 by six region-based inspectors, the senior resident inspector, and a section chief. The inspection covered site project management; quality assurance; design control; welding; electrical supply and distribution activities; nondestructive examination; structural installations; and mechanical installations. The inspection focused on hardware associated with the High Pressure Core Spray system. Two potential violations were identified; one for particulate contamination of piping systems and another for undersized welds on an instrumentation support rack. The inspection noted an improvement in the conduct of site quality activities post - I&E CAT inspection and found the installed hardware to meet the regulatory criteria with the exceptions identified above as potential violations. The inspection report has not yet been formally issued, and the licensee will be required to respond to the noted concerns.

3.4 Independent Non-Destructive Examination (NDE)

An independent Non-Destructive Evaluation (NDE) verification by NRC Region I was conducted during Spring 1984 using the Region's mobile NDE laboratory. The inspection involved 662 inspection hours.

The purpose of the inspection was to verify the adequacy of the licensee's quality control program for NDE through independent testing. This was accomplished by performing the same tests that the licensee had performed, and then comparing Region I results to those of the licensee. The program also performed pipe wall thickness measurements and radiographic film comparison.

An NRC inspector made a random selection of weldments. These were intended to provide a representative sample of piping systems, components, and structural weldments which represent various pipe sizes, shop and field weldments fabricated to AWS and ASME Class 1, 2, and 3 codes. The items selected were previously accepted by the licensee based on vendor shop and on site QA/QC records.

The Region I examinations were performed using detailed procedures specifically written for compliance with the licensee's PSAR commitments to the ASME III Boiler and Pressure Vessel Code. The intent was to duplicate, to the extent practicable, the techniques and methods of the original examinations. The results of those independent examinations were as follows:

Radiography (RT) - Thirty-five welds were examined by radiography using an Iridium-192 source. The weld sample included ASME III Class 1,2,and 3 carbon and stainless steels. All welds were found acceptable.

Liquid Penetrant (PT) - Thirty welds and the adjacent base metal were examined by liquid penetrant. All areas examined were found acceptable.

Magnetic Particle Examination (MT) - Thirty ASME III pipe welds and AWS structural welds were examined using magnetic particle techniques. All areas examined were acceptable.

Visual Examination (VT) - Eighty-six weldments and adjacent base material were visually inspected for weld reinforcement, overall workmanship, and surface condition. Forty-one ASME NF pipe support welds and twenty AWS D1.1 structural welds were examined. All areas examined were acceptable.

Thickness Measurement - Thirty-four welds and adjacent pipe material were examined using an ultrasonic thickness gauge. Minimum wall thicknesses were determined from ASTM standard pipe size and nominal thickness charts. One ASME shop weld was found to be 0.015 inches below minimum wall thickness and a violation was issued.

Anchor Bolt Ultrasonic Examination - Forty installed concrete anchor bolts were ultrasonically examined for proper length, all were found to be acceptable.

Hardness Measurements - Twenty-six welds were examined for Brinnell hardness values, all were found to be acceptable.

Radiographic Film Review - One hundred and sixty-eight pipe weld radiographs were reviewed to verify accurate film interpretation and the adequacy of the radiographic program. One unacceptable linear indication was identified for which an interpretation was not recorded on the reader sheet and another film was found to portray an unacceptable transverse linear indication. Two violations were issued.

The Region I independent NDE verification showed generally good agreement with the applicant's determinations. Further licensee review of radiographic film has been accomplished to provide assurance of the technical adequacy of the film interpretations.

3.5 Inspection and Enforcement Construction Appraisal Team (CAT) Inspection

The Office of Inspection and Enforcement (I&E) conducts Construction Appraisal Team (CAT) inspections to evaluate the implementation of management control of construction activities at selected facilities and to evaluate the quality of construction at nuclear plants. The inspection team was comprised to six NRC inspectors; five consultants; and a team leader. The inspection consisted of a detailed examination of selected installed hardware subsequent to the performance of licensee quality control inspections; a selective examination of procedures and records; observation of in-process work; and interviews of site personnel. The CAT inspectors examined the areas of electrical and instrumentation installations; mechanical construction; welding installations; nondestructive examination reviews; structural construction; material traceability, equipment storage and maintenance; and Quality Assurance activities.

The inspection detected a broad range of problems in the construction and inspection programs as identified below:

Electrical and Instrumentation Construction

Electrical cable and raceway installations exhibited separation problems, particularly in the control room complex, that had not been identified by Quality Control nor had the licensee defined how these deficiencies would be rectified at a later date. Indeterminate bolting material was utilized on the station battery racks and at shipping splits for switchgear and motor control center enclosures. Inspection records did not reflect the design documents to which the inspection had been performed. The inspection procedures were found deficient with respect to attributes on raceway marking, acceptance criteria for separation, bolting material identification, and protrusions into cable trays.

Mechanical Construction

The inspection found that HVAC and piping runs were generally erected in accordance with the applicable requirements. Several pipe supports were found to contain deficiencies not previously identified by contractor QC personnel. The concrete expansion anchor installation program acceptability was questioned as the pre-qualification tests had been performed in a

concrete mix of lower compressive strength than normally used for seismic Category I structures. Significant loss of pre-load was observed when the installed expansion anchors were torque tested.

Welding and Nondestructive Examination

The NRC Construction Appraisal Team inspection identified major deficiencies in the ITT Grinnell program for ASME code radiographs. The associated Stone & Webster Engineering Corportaion and the applicant's program for review and acceptance of these radiographs was found to be deficient. Problems involving weld quality, film quality, and inadequate documentation were identified by the NRC Construction Appraisal Team. It was noted that some similar problems were also identified by site quality assurance/quality control programs; yet timely corrective actions were not being taken. Inadequate liquid penetrant surface examination of ASME pressure boundary welds was identified. The site structural welding to the AWS Code was found to be generally satisfactory.

Civil and Structural Construction

The concrete and structural steel installations were found to be in conformance with regulatory and specification requirements. Minor problems were identified with concrete placement activities and records.

Material Traceability, Storage and Maintenance

The project storage and maintenance programs were examined and found to be acceptable. Inadequate control of unused weld filler material was identified and some pipe support members lacked the requisite material traceability.

Quality Assurance

The inspection findings indicated that the licensee had implemented an ineffective audit program. Nonconforming items were found to have been dispositioned on documents other than formal QA documents such that the trend programs were bypassed. Inadequate document control measures were observed in that out of date drawings were utilized in the field and inspection records were found to not reflect the design documents used during the conduct of the inspection. Untimely conduct of Quality Control inspections was observed due to the wide disparity between the number of items completed by construction in relationship to those inspected by Quality Control.

On March 20, 1984 an Enforcement Action (EA) was issued predicated upon the CAT identified problems at Nine Mile Point 2. The Enforcement was composed of a Notice of Violation, an Order, and a proposed Civil Penality of $100,000. The Notice of Violation reiterated the aforementioned CAT deficiencies. The Order called for an independent review of site corrective action programs, the development of a site management quality performance trend program, and the conduct of a management review audit.

The licensee responded to the Enforcement Action with steps to address the specific deficiencies and provided corrective/preventive action plans. A consultant (Management Analysis Company) was retained by the licensee to perform the review of corrective action programs. The licensee committed to develop a site quality trend program that would serve to monitor the quality levels of various installation activities.

In response to the CAT identified deficiencies, the licensee instituted a complete management reorganization at both corporate offices and the site. The following Niagara Mohawk personnel were removed from their involvement with Nine Mile: Senior Vice President; Vice President Nuclear Construction; Vice President of Quality Assurance; Construction QA Manager; Construction Manager; and Site QA Supervisor. The licensee has retained Management Analysis Company (MAC) to provide key nuclear experienced managers to fill both project and quality assurance positions. The licensee has significantly improved their control of site activities. Project management is located on-site and the line organization is such that the Stone and Webster project director now reports directly to the licensee project director. The QA managers of Stone and Webster and subcontractors have either been replaced or augmented with additional corporate level QA management to enhance the effectiveness of the QA organizations.

In addition to the management reorganization outlined above, the following correct actions have been implemented over the course of the past year

-- Increased licensee surveillance/audits of contractor performance, particularly hardware related.

-- Licensee QA review of inspection procedures/attributes to ensure adequate accept/reject criteria definition.

-- Sampling reinspection of mechanical equipment bolting to verify acceptability.

-- Sampling reinspection of structural steel weldments.

-- The Power Generating Control Complex (PGCC) was inspected to identify electrical separation violations.

-- The licensee performed in-depth assessments of the contractor QA organizations.

-- Hardware reinspections of previously accepted installations to establish the quality level of in site hardware.

-- The piping contractor re-performed liquid penetrant tests on ASME pressure boundary welds in response to the CAT identification of deficient weld examinations.

-- ASME pipe weld radiographs were re-interpreted to ensure that all rejectable indications had been resolved as corrective action to CAT identified radiographic violations.

Region I reviewed the CAT report and has identified the items that were classified as Violations or Followup items. These issues are currently tracked as open items. The licensee has implemented numerous corrective actions to address the problems in the form of hardware reinspections, radiographic film reviews; re-performance of nondestructive examinations, testing of concrete expansion anchor bolt to demonstrate load carrying capacity, enhancement of quality programs; increased audit/surveillance conduct, and management personnel reorganization. Region I has closely monitored the implemented actions, however, the majority of the CAT findings are still identified as open concerns pending formal Region I verification of implemented actions and closure within an inspection report.

The licensee developed a Quality Performance Management Program (QPMP) which monitors the quality status of the site. Key parameters such as quantity installed, quantity inspected, and QC acceptance rates are monitored for construction hardware commodities. The program monitors outstanding design changes and open QA deficiency documents. Trending is performed on some of the documented nonconforming conditions. Region I is monitoring the utilization of QPMP by the licensee through management meetings in conjunction with review of QPMP data; and attendance at the QPMP licensee review meetings.

3.6 Review of Construction Deficiencies

Significant deficiencies in design and construction, as defined in 10 CFR 50.55(e), are required to be reported to the NRC. Licensee management attention, has been devoted to this reporting activity and the licensee has been responsive to the 10 CFR 50.55(e) requirements.

Review by Region I indicates the licensee program for significant deficiency reporting is appropriately implemented. Nonconformances and programmatic QA deficiencies are reviewed by Quality Assurance and engineering personnel to determine whether items meet the criteria for reportability. Licensee upper management from QA, construction, operations, and engineering compose a site review committee to further evaluate deficiencies for reportability.

As of January 4, 1985, the licensee had reported 117 significant deficiencies, of which 29 are resolved and 88 remain outstanding. Licensee corrective actions have been instituted in response to the reported deficiencies, however the licensee has yet to verify the corrective actions. The items will be subsequently reviewed by Region I inspectors for closure.

The reported deficiencies for Nine Mile Point 2 have involved a variety of problems. The following item is an example of the licensee identification and correction of a 10 CFR 50.55(e) item:

CDR #82-00-02 - Defective Emergency Diesel Generator Lube Oil Strainer Baskets

The licensee was notifed by Cooper Energy Services via a 10 CFR 21 report that a design deficiency existed with the diesel generator lube oil strainer baskets. Performance testing of the diesel by Cooper had demonstrated that the mesh strainer basket liner was torn loose at the top of the basket. Further testing showed the strainer mesh to disintegrate, with the potential to cause a bearing failure thus disabling the diesel generator.

The strainer baskets were redesigned and tested by the diesel manufacturer with no observed problems. SWEC initiated Nonconformance and Disposition (N&D) Report 4923 which reiterated the Part 21 information. The N&D was found to be a reportable item. Region I was verbally notified on February 5, 1982 of the deficiency with a followup report on March 4, 1982. The diesel vendor supplied the new strainer baskets, with installation directions, to the site. The new baskets were receipt inspected via Inspection Report X4002575 and Procurement Quality Assurance certifications were provided. The defective liner was replaced and SWEC QC verified the reassembly of the lube oil strainers. The corrective actions were reviewed by Region I and found to be acceptable as documented within Inspection Report 50-410/84-15.

3.7 Followup on Allegations

Allegations received by NRC Region I addressed both safety-related and non-safety-related areas. Each allegation is reviewed by regional management, and appropriate followup is determined based on potential safety significance. Significant allegations are investigated by members of the NRC Office of Investigations.

Region I records indicate that 25 Nine Mile Point 2 project allegations have been received. Of those allegations which have been substantiated, 3 resulted in enforcement actions as noted below:

- An allegation was received in February, 1983 that ITT welders were not provided weld procedures for ASME piping installation. The concern was substantiated through interviews with craft supervision and rod room attendants and a Level IV violation was issued. The licensee conducted extensive training programs to ensure that craft personnel were aware that weld procedures are maintained for reference at all ITT weld rod issue stations. The licensee verified that the weld procedures were indexed at the field stations.

- An allegation was received in April 1983 that inconsistent diesel generator loading sequences were assumed by General Electric and Stone and Webster Engineering Corporation. The FSAR was found to be inaccurate in regards to the load sequencing. A Level V violation was issued with regard to inaccurate design review. The licensee amended the FSAR to portray the correct load sequencing and instituted measures to enhance the review of FSAR submittals.

- An allegation was received in September 1982 that uncertified quality control personnel were performing inspections. Subsequent Region I investigation verified that inspections of safety-related electrical equipment had been performed solely by uncertified trainees. The associated inspection records were falsified, as qualified Level II inspectors had signed the documents, which signified the inspections were performed by certified personnel. The deficient inspection conduct and falsified inspection records demonstrated the need for the licensee to increase their management involvement with contractor activities to assure proper implementation of site QA/QC programs. The licensee reinspected the work performed by the trainees and did not identify substantive deficiencies.

The licensee has recently established an allegation clearinghouse. The program is identified as the Quality First Program (Q1P). Site employees are encouraged to report any quality concerns so that appropriate investigations can be performed. The Q1P program conducts exit interviews and maintains a toll free telephone line to report concerns. The QIP program provides feedback to the concerned employee after the investigation is completed.

3.8 Systematic Assessment of Licensee Performance (SALP)

The SALP program encompasses an integrated NRC staff review of licensee performance over an annual or eighteen month cycle. An analysis is performed with regards to the NRC observed strong points and weaknesses of the licensee construction and quality assurance efforts. The process serves to identify those areas to which licensee management should devote greater attention and to which greater NRC inspection resources will be allocated to achieve benefits in the quality of facility construction. Since the inception of the SALP program, the performance of the licensee has been assessed three times.

The first SALP addressed performance during the period from February 1, 1980 to January 31, 1981. Inspection activity had covered the containment structure, reactor pressure vessel installation, reactor pressure vessel internals, biological shield wall, and quality assurance. Only one notice of violation had been issued during that time frame for inadequate incorporation of design changes into the associated engineering drawings. The licensee had reported four non-causally linked 10 CFR 50.55(e) construction deficiency reports. No change was found necessary for the implementation of the Region I inspection program.

The second SALP addressed performance during the period from October 1, 1981 to September 30, 1982. The licensee performance was assessed as follows:

Functional Area	Category 1	Category 2	Category 3
Soils and Foundations; Containment and Other Safety-Related Structures	X		
Piping Systems and Supports; Safety-Related Components; Support Systems; Electrical Power Supply; Licensing		X	
Quality Assurance			X

Close licensee surveillance of the Soils and Foundation area contributed to the fact that no safety concerns were identified. A singular violation in regard to steel reinforcing bar substitution was the only adverse finding in the Safety-Related Structures area. A violation was detected wherein ASME Class I weld procedures had not been properly qualified for impact testing. Two enforcement conferences (50-410/82-06 and 50-410/82-08) had been held in June and July 1982 to discuss this particular problem. Region I attributed the deficiency to the following root causes: inexperience of the piping contractor personnel; ineffective Stone and Webster control of the piping contractor; and inadequate licensee attention to sub-contractor activities. The licensee committed to strengthen the piping contractor's QA and engineering staffs; increase Stone and Webster audits of the piping contractor; to increase management involvement with contractor activities; and to reorganize the licensee corporate offices to create a new position of Vice President - QA and Senior Vice President Nuclear Services to serve as responsible licensee management for all project matters, Region I identified deficiencies in the conduct of Preventive Maintenance activities resulted in the assumption of those responsibilities by Stone and Webster. Several concerns with the licensee implementation of the HVAC inspection program were raised. Four violations were identified in the Electrical Power Supply area including failure to specify design separation criteria in the installation specification; failure to assure that cable trays conform to procurement specifications; incorrect design of cable tray weldments; and inadequate welds deposited on cable tray cross braces. Significant weaknesses were identified within licensee an contractor QA programs including: inexperienced personnel on the licensee, Stone and Webster, and piping contractor QA staffs; failure of QA to identify programmatic weaknesses; licensee overdependence upon Stone and Webster QA to assure plant quality; inability of the licensee to resolve audit findings; lack of licensee control of Stone and Webster; and inadequate licensee management involvement in the project. The licensee responded to the identified deficiencies by implementing the following actions: creating a position of Vice President for Quality Assurance; adding personnel to the site QA staff; requiring increased licensee management presence onsite; increased licensee QA surveillance conduct; increased Stone and Webster Surveillance of contractor activities; and increased overall licensee management control of Stone and Webster activities.

The third SALP addressed performance from October 1, 1982 to September 30, 1983. The licensee performance was assessed as follows:

Functional Area	Category 1	Category 2	Category 3
Soils and Foundations	X		
Containment and Other Safety-Related Structures; Safety-Related Components; Electrical Power Supply and Distribution; Instrumentation and Control Systems; Licensing Activities		X	
Piping Systems and Supports			X
Project Management/Quality Assurance			

Region I identified two violations regarding the QC acceptance of non-conforming concrete expansion anchor bolts and the inadequate QC inspection of structural steel bolted connections. The licensee performed re-inspections to assure the adequacy of other hardware. Concrete placement activities were observed to be properly conducted. Major deficiencies were identified by the licensee in the conduct of the piping contractor's radiography program. Several instances of film enhancement were discovered. The questionable welds were re-radiographed and found acceptable. Problems with the piping contractor control of field installations demonstrated that the staff additions implemented previously by the licensee were not totally effective. Three violations were identified in the electrical area on design control of the Standby Diesel Generator load sequencing; acceptance of equipment with deficient vendor internal wiring; and inadequate inspection of partial pulled cable installations, Further problems were identified that illustrate greater licensee attention is necessary to ensure the piping contractor performance meets the requisite quality criteria. The licensee project management direction and contractor oversight functions were found to be lacking as illustrated by the following deficiencies: inadequate QA review of weld planner packages; lack of control over Nondestructive Examination process; use of Stone and Webster trainees to conduct inspections; lack of non-conformance trend analysis by sub-contractor, piping contractor QA management made statements to restrict ability of personnel to express quality concerns to the NRC. An enforcement conference (50-410/82-13) was held on October 20, 1982 to discuss the improper use of QC trainees and falsification of inspection records. Region I expressed that greater licensee overview is required of project QA activities. The licensee committed to increased site QA staffing levels and greater QA management attention. On May 23, 1983 a Management Meeting (50-410/83-09) was held to again discuss the necessity for increased licensee control over the piping contractor. Another Enforcement Conference (50-410/83-14) was convened on August 30, 1983 to discuss the piping subcontractor prohibition on personnel from bringing safety concerns to

the NRC. Steps were taken to increase the piping contractor QA staffing and to increase the licensee surveillance of their contractor's activities. The licensee instituted major organization changes as described in section 3.5 of this report to enhance their control of project activiites.

A region I evaluation board is scheduled to convene on March 18, 1985 to assess the fourth SALP period from October 1, 1983 to January 31, 1985.

3.9 Region I Overview

The early inspections conducted at Nine Mile Point 2 through late 1981 did not identify substantive deficiencies. The work activities involved site preparation, concrete placement, RPV placement, and QA program reviews. While several violations and/or infractions were issued, no programmatic problems were apparent.

The Regional Construction Team Inspection conducted in November, 1981, in concert with the assignment of a site resident inspector, served to identify further deficiencies that were not previously apparent. The concentrated team inspection approach found several problems with the mis-application of site QA programs and the inadequate program implemented by the licensee to monitor their contractors. The resident inspector identified deficiencies within the piping contractor's control of welding activities that resulted in two enforcement conferences between Region I and the licensee. The licensee was informed that greater management attention and surveillances needed to be performed to ensure control of the piping contractor.

Close Region I scrutiny of the site QA programs resulted in the identification of Stone and Webster use of trainees to perform inspections and the associated falsification of inspection records. Further problems indicated that licensee steps to shore up the performance of the piping contractor had not been fully effective. Between October 1982 and August 1983, three more Management meetings/Enforcement conferences were held with the licensee. Region I emphasized throughout, that the licensee needed to implement more effective control of, and perform enhanced surveillance to ensure that the subcontractors adequately performed their work.

The I&E CAT inspection in late 1983, served to bring to a culmination, the fact that the site wide QA programs were deficient.

The licensee subsequently instituted major management reorganizations to bring experienced personnel on-site to guide the remainder of construction activity. Site software, inspection procedures, have been enhanced to closely identify inspection criteria. Numerous reinspections have been performed of plant hardware to assure its adequacy relative to the design criteria. Results of Nondestructive Examination activity have been reviewed to assure pressure boundary integrity. The new project management has instituted changes in both construction and QA activity to enhance the quality of site construction.

Region I continues to inspect on-going work activities, review the adequacy of licensee corrective actions to previously identified deficiencies, monitor actions taken in response to the CAT enforcement action; and will inspect the conduct of preoperational testing. These actions will serve to provide the data base necessary to judge the future licensibility of the Nine Mile 2 project.

3.10 Preoperational Testing

No safety-related systems have been turned over to the Startup and Test Group for preoperational testing. Some safety-related equipment has been released to NMPC for preliminary testing such as flushing. The reactor vessel is scheduled to be filled February 22, 1985. Following an integrated flush, the reactor vessel is scheduled for hydrostatic testing March 18, 1985.

The licensee has established the administrative controls for conducting the Startup and Test Program. Region I has reviewed them and found no unacceptable condition. A resident inspector dedicated to preoperational testing will be on site full-time March 15, 1985.

4.0 Conclusion

Region I finds that the current construction quality program to be generally acceptable. Recent inspection activities continue to identify problems and the licensee will have to institute corrective actions in response to those identified problems. Region I has an extensive list of outstanding problems for which, licensee corrective actions have been instituted, but remain to be verified and ultimately closed out as acceptable. The licensee has assigned experienced quality assurance and project management personnel to resolve site issues. Region I will monitor the remaining phases of construction and pre-operational activity as well as determining the acceptability of previously installed hardware to ensure that regulatory requirements are fulfilled.

NINE MILE POINT UNIT 2

INSPECTION

SUMMARY

INSPECTION NUMBER	INSPECTION DATE	NUMBER OF INSPECTORS	AREAS INSPECTED	FINDINGS	VIOLATIONS
72-01	4/10/72	2	Initial management meeting	None	None
72-02	7/28 - 8/24/72	1	QA Program	3 unresolved items	None
73-01	6/14/73	2	QA Program	None	None
73-02	7/19 - 10/9/73	1	QA Program	None	None
74-01	9/26 - 9/27/74	1	QA Program	6 unresolved items	None
75-01	4/16 - 4/17/75	2	QA Program for procurement, and surveillance review	None	None
75-02	7/9 - 7/10/75	1	Excavation progress, engineering review/ approval program, review of construction procedures	2 unresolved items	None
75-03	7/29-8/1/75	1	Environmental protection program	None	None
75-04	10/21 - 10/22/75	2	Site preparation activities and procedure review	2 unresolved items	None
76-01	2/24 - 2/26/75	2	Specification review, review of audits, material storage, corrective action controls, review of project manual	2 unresolved items	Infraction: Requisite QA requirements not identified in specification. Infraction: Specification review not performed in accordance with procedures. Deficiency: Audit findings not properly documented.
76-02	4/12 - 4/14/76	2	Blasting records, QA procedures for concrete and foundations, site preparation, groundwater control	4 unresolved items	None

A-77

INSPECTION NUMBER	INSPECTION DATE	NUMBER OF INSPECTORS	AREAS INSPECTED	FINDINGS	VIOLATIONS
76-03	5/25 - 5/27/76	2	Site preparation excavation mapping, design review meetings, porous concrete foundation drainage system, batch plant qualification, concrete mixes	3 unresolved items	None
76-04	7/19 - 7/21/76	1	Concrete activities, batch plant operation	None	Infraction: Inadequate cadweld inspections
76-05	8/16 - 8/20/76	1	Environmental protection activities	1 unresolved items	None
76-06	9/28 - 9/30/76	1	Foundation backfill operations, site preparation	3 unresolved items	Infraction: Concrete curing inspections not performed
76-07	11/16 - 11/18/76	1	Site preparation and foundation records, QC personnel qualification, batch plant operation	1 unresolved	Infraction: Uncon- repair of cadwelds
77-01	3/22 - 3/24/77	1	Containment base mat concrete, geologic investigation on rock stress	None	None
77-02	4/5 - 4/7/77	2	Containment reinforcing steel cadwelding, QA Manual review for containment liner contractor	2 unresolved items	None
77-03	4/14/77	1	Containment steel liner installation, geologic investigation review	None	None
77-04	5/2 - 5/5/77	2	Containment structural steel, containment concrete procedures, containment base mat concrete activities, nonconformance program review	2 unresolved items	None
77-05	5/17 - 5/19/77	1	Containment base mat concrete, containment liner welding	1 followup item	None

A-78

INSPECTION NUMBER	INSPECTION DATE	NUMBER OF INSPECTORS	AREAS INSPECTED	FINDINGS	VIOLATIONS
77-06	6/8 - 6/10/77	1	Containment base mat concrete, containment structural steel welding, component storage, reinforcing steel erection	3 unresolved items	None
77-07	6/15/77	2	Containment base mat concrete	None	None
77-08	7/18 - 7/20/77	1	Primary containment steel liner welding, control of weld material, weld inspection	None	None
77-09	8/3 - 8/5/77	1	Environmental protection program	None	Infraction: Settling pond operation procedures not followed.
77-10	8/23 - 8/25/77	2	Component storage and maintenance, containment base mat concrete records	1 unresolved	Deficiency: Failure to perform proper preventive maintenance on recirculation pump motors.
77-11	10/11-10/13/77	3	QA record review for concrete base mat concrete	1 Followup	Infraction: Failure to follow procedures for batch plant uniformity tests Infraction: Failure to document and report corrective actions to management relative to I&E Bulletins and Circulars
77-12	11/15 - 11/17/77	2	Installation and welding of containment liner	1 unresolved item	None

A-79

INSPECTION NUMBER	INSPECTION DATE	NUMBER OF INSPECTORS	AREAS INSPECTED	FINDINGS	VIOLATIONS
78-01	1/31 - 2/2/78	1	Measuring and test equipment non-destructive weld examination	None	None
78-02	3/28 - 3/31/78	1	Reinforcing bar installation, I&E Bulletins and Circulars	None	None
78-03	4/24 - 4/26/78	3	Piping and supports installation, penetrations, geotechnical review	None	None
78-04	5/15 - 5/18/78	2	Equipment storage, surveillances, concrete activities	None	None
78-05	6/14/78	1	Cable tray support records	None	None
78-06	7/25 - 7/27/78	1	Mechanical equipment installation, equipment maintenance, I&E Bulletins and Circulars	1 unresolved item	None
78-07	9/26 - 9/28/78	1	Risk release program, equipment installation documentation	1 unresolved item	None
78-08	9/25 - 9/29/78	1	Management controls	2 unresolved items	None
78-09	10/30 - 11/2/78	1	Concrete activities, equipment installation	1 unresolved item	None
79-01	1/8 - 1/11/79	1	Structural steel erection, cad weld operations	2 unresolved items	None
79-02	3/20 - 3/23/79	2	Containment liner welding, structural steel erection, stud welding, equipment storage	None	Infraction: Required radiography examination not performed.
79-03	4/24 - 4/26/79	1	Reinforcing bar,concrete activities, backfill operation	1 unresolved item	None
79-04	7/10 - 7/12/79	1	Containment liner installation, piping, structural steel erection	None	None

INSPECTION NUMBER	INSPECTION DATE	NUMBER OF INSPECTORS	AREAS INSPECTED	FINDINGS	VIOLATIONS
79-05	5/14 - 5/17/79	1	Structural steel erection, containment liner installation	None	Infraction: No procedure for ultrasonic examination
79-06	8/27 - 8/31/79	1	Concrete activities, reinforcing bar storage, electrical installation	1 unresolved item 1 followup item	None
79-07	10/9- 10/12/79	2	Suppression pool downcomers, electrical activities	4 unresolved items 2 followup items	None
79-08	11/6 - 11/8/79	1	Reactor pressure vessel transport/ rigging/lifting	None	None
79-09	11/28 - 11/30/79	2	Containment liner welding, equipment rigging, valve installation	1 unresolved item	None
80-01	1/9 - 11/11/80	1	Equipment maintenance and storage	1 unresolved item	Infraction: Invalid preventive maintenance records
80-02	3/19 - 3/21/80 & 4/10 - 4/11/80	3	Reactor pressure vessel storage, biological shield wall installation	None	None
80-03	4/24/80	2	Plant Tours	None	None
80-04	5/12 - 5/15/80	2	Repair of biological shield walls welds	2 unresolved items	None
80-05	6/17 - 6/19/80	1	Containment reinforcing bar cadweld operations	2 unresolved items	None
80-06	7/14 - 7/18/80	1	Reactor pressure vessel storage and placement	None	None
80-07	7/15 - 7/18/80	2	QA procedure review, review of engineering design changes, nonconformance report disposition, trend analysis, audit program	1 unresolved	None

A-4

INSPECTION NUMBER	INSPECTION DATE	NUMBER OF INSPECTORS	AREAS INSPECTED	FINDINGS	VIOLATIONS
80-08	8/4 - 8/8/80	2	Reactor pressure vessel placement	None	None
80-09	9/9 - 9/11/80	1	QC record review, reactor pressure vessel storage, CRD hydraulic control unit storage	1 unresolved item	None
80-10	9/23 - 9/26/80	1	QA Program review	None	None
80-11	10/21 - 10/23/80 & 11/4 - 11/6/80	1	Concrete record documentation review	None	Infraction: Inadequate document control of engineering changes.
81-01	1/20 - 1/23/81	1	Concrete placement, reinforcing bar cadwelds, engineering change document review	1 unresolved item	None
81-02	2/18 - 2/25/81	1	Environmental protection program	None	None
81-03	4/21 - 4/23/81	1	Plant tour, cadweld operations	None	None
81-04	4/22/81	2	SALP management meeting	None	None
81-05	6/23 - 6/25/81	1	Primary containment liner, equipment storage, housekeeping	3 unresolved items	None
81-06	7/14 - 7/16/81	1	I&E Bulletin and Circular review	None	None
81-07	7/27 - 7/31/81	2	Electrical procedure review, installed raceway, geologic fault study review, primary containment concrete	1 unresolved item	Violation: Pump motor electrical test data not submitted to engineering
81-08	8/4 - 8/6/81	1	QA program personnel certifications, licensee audits	None	None
81-09	8/18 - 8/21/81	3	Primary containment record review, biological shield wall welding, ROV records, weld filler metal control, pipe welding	1 unresolved	None

A-62

INSPECTION NUMBER	INSPECTION DATE	NUMBER OF INSPECTORS	AREAS INSPECTED	FINDINGS	VIOLATIONS
81-10	9/1 - 9/3/81	2	Plant tour, concrete aggregates, corrective action programs	1 unresolved	None
81-11	9/29 - 9/30/81	3	Plant tour, review of QA organization	None	None
81-12	10/13 - 11/13/81	1	Containment penetrations, piping and structural steel erection	1 unresolved item 1 followup item	Violation: Incorrect weld data sheet Violation: Inadequate corrective actions
81-13	11/30 - 12/18/81	5	Region I CAT inspection, welding, non-destructive examination, electrical, structural, procurement	1 unresolved item	Violation: Ineffective QA program Violation: Lack of design control Violation: Inadequate equipment qualification
81-14	12/21/81-1/15/82	1	Structural steel erection, piping activities, welder qualification	1 unresolved item 1 followup item	Violation: Failure to implement checklists
82-01	1/18 - 2/26/82	1	Pipe whip restraints, structural steel, welder qualification, nondestructive examination, piping, procurement	3 unresolved items 2 followup	Violation: Inadequate QC personnel training Violation: Failure to follow instructions
82-02	3/1 - 3/26/82	1	Pipe supports, RPV nozzle modifications, structural steel, receipt inspection, nondestructive examinations, piping	3 unresolved items	Violation: Failure to record data in accordance with instructions.
82-03	3/29 - 4/30/82	1	Structural steel, cadwelding, expansion anchors, piping, equipment, design control	4 unresolved items 4 followup items	Violation: Failure to to correctly translate design information Violation: Failure to impose QA requirements on purchase orders

A-63

INSPECTION REPORT	INSPECTION DATE	NUMBER OF INSPECTORS	AREAS INSPECTED	FINDINGS	VIOLATIONS
82-04	5/11 - 5/13/82	1	Electrical components	3 unresolved items	None
82-05	5/10 - 6/3/82	1	Piping, HVAC installation, structural steel	1 followup item	None
82-06	6/1/82	4	Enforcement conference on ITT Class 1 weld procedures	None	None
82-07	6/21 - 7/23/82	1	Piping, rigging, mechanical equipment storage, structural steel	1 unresolved item 4 followup items	Violation: Inadequate control of rigging operations Violation: Failure to identify a nonconforming condition.
82-08	7/16/82	5	Meeting on ITT welding procedures	None	Violation: Failure to control welding processes
82-09	7/13 - 7/16/82 & 7/20/82	2	Management controls	1 unresolved item	None
82-10	7/26 - 8/27/82	1	Concrete testing, concrete expansion anchors, piping, reactor head cavity, pit, review of engineering design changes	2 unresolved items	Violation: Underlength welds on cable tray braces Violation: Aggregate material not properly tested. Violation: Inadequate inspection of concrete expansion anchors
82-11	8/30 - 9/30/82	2	Mechanical equipment erection, piping diesel generator building, weld rod control, welder qualification, pipe supports, raceway	4 unresolved items 2 followup items	Violation: Work performed without planner package Violation: Weld filler material not properly stored

A-54

INSPECTION NUMBER	INSPECTION DATE	NUMBER OF INSPECTORS	AREAS INSPECTED	FINDINGS	VIOLATIONS
82-12	10/12 - 11/12/82	1	Small bore piping, instrumentation, structural steel, recirculation pipe welds, nonconformance reports	2 unresolved items 2 followup items	Violation: ASME piping planner package not properly reviewed
82-13	10/20/82	1	Enforcement conference regarding use of trainees to conduct inspection	None	None
82-14	11/15 - 12/22/82	1	High strength bolting, electrical support welds, structural steel welder qualifications, measuring and test equipment	1 unresolved item 2 followup items	Violation: High strength bolts not inspected per AISC Violation: Weld material requisition improperly filled out Violation: Improper concrete anchor bolt installations.
82-15	12/13 - 12/17/82	1	Bioshield wall fill material placement reactor building concrete activities	None	None
82-16	12/14 - 12/16/82	1	Equipment storage, raceway installation	2 unresolved items	None
83-01	1/3 - 2/4/83	2	Instrumentation, pipe whip restraints, engineering change documents, pipe supports, piping, cables	4 unresolved items 7 followup items	Violation: Weld procedures not available to welders
83-02	2/7 - 3/11/83	1	Safety related equipment, concrete installation, cables, structural welding, piping	3 unresolved items 2 followup items	Violation: Planner package not adhered to for pipe installation activities
83-03	3/1 - 3/3/83	1	Electrical equipment storage raceway installation	5 unresolved items	None

A-85

INSPECTION REPORT	INSPECTION DATE	NUMBER OF INSPECTORS	AREAS INSPECTED	FINDINGS	VIOLATIONS
83-04	3/14 - 4/15/83	1	Piping, licensee surveillances and audits, instrumentation supports, structural steel, HVAC duct	2 unresolved items 1 followup item	None
83-05	4/25 - 5/27/83	2	Recirculation piping, containment liner, procurement, corrective action, instrumentation supports, piping and pipe supports	1 unresolved item 4 followup items	Violation: Inadequate trend analysis Violation: Underlength fillet welds
83-06	5/16 - 6/1/83	1	QA record review, design verification procedures, as-built plant verification	2 unresolved items	Violation: Inconsistent FSAR diesel generator load sequencing
83-07	6/13 - 6/17/83 & 7/13 - 8/5/83	1	Pipe supports, piping, structural steel, weld filler material control, cable and equipment installation	3 unresolved items 3 followup items	None
83-08	6/7 - 6/9/83	1	Switchgear installation and instrumentation	3 unresolved items	Violation: Deficient vendor wiring on electrical equipment
83-09	5/23/83	1	Management meeting regarding licensee overview of contractor performance	None	None
83-10	7/25 - 7/29/83	2	Corrective action programs, piping and plant housekeeping	2 unresolved items 1 followup item	None
83-11	8/2 - 8/4/83	1	Cables, raceway, motor control centers, electrical QA records	1 unresolved item 1 followup item	None

INSPECTION NUMBER	INSPECTION DATE	NUMBER OF INSPECTORS	AREAS INSPECTED	FINDINGS	VIOLATIONS
83-12	8/8 - 9/21/83	2	Preventive maintenance, pipe supports, cables, concrete placement, recirculation piping nozzle modification	3 unresolved items 5 followup items	Violation: Installed cable not properly inspected Deviation: Inspection hold points not specified in QC inspection plans
83-13	9/14/83	2	Environmental protection program	None	None
83-14	8/30/83	1	Enforcement conference on licensee overview and investigations of the piping contractor	None	None
83-15	10/18 - 10/20/83	1	Electrical cables, motor control centers and QA records	1 unresolved item	None
83-16	10/1 - 12/2/83	1	Equipment turnover, piping, pipe supports, reactor building enclosure, CRD piping, fire protection, instrumentation QA program	6 unresolved items 2 followup items	Violation: Nonconforming welds accepted by quality control
83-17	12/5/83 - 1/20/84	2	RPV storage, hydraulic control unit installation, piping, pipe supports, welder qualification, HVAC systems	4 unresolved items 5 followup items	None
83-18	11/7 - 11/19/83 & 11/28 - 12/9/83	7	I&E CAT inspection	None	Violation: Inadequate review of design change documents, Inadequate review of radiographic film, deficient inspection procedures, nonconforming pipe supports, electrical separation violations, Inadequate inspection documentation, Inadequate weld filler metal control, Deficient NDE weld surface exams, nonconforming conditions not identified on NCRs, Inadequate corrective action programs, Ineffective audit programs.

4-4

INSPECTION NUMBER	INSPECTION DATE	NUMBER OF INSPECTORS	AREAS INSPECTED	FINDINGS	VIOLATIONS
84-01	1/23 - 3/2/84	2	Pipe supports, diesel generator cranes, reactor vessel internals, QA program	4 unresolved items 1 followup item	Violation: Nonconforming pipe supports Violation: Failure to follow procedure for handling of significant deficiency
84-02	2/7 - 2/9/84	1	Cables, switchgear and QA records	5 unresolved items	Violation: Wrong nonconformance form in use
84-03	2/22/84	1	Management meeting on licensee corrective actions for CAT findings	None	None
84-04	3/12 - 3/16/84	1	Concrete anchor bolts and structural steel welding	None	None
84-05	3/5 - 4/7/84	1	Electrical terminations, piping, pipe supports, QA surveillances, contractor audits, design control of II/I items	3 unresolved items 1 followup item	Violation: Inadequate design control for Seismic II/I items
84-06	4/9 - 5/11/84	2	Structural steel, weld material control, pipe supports, housekeeping, post inspection rework control	2 unresolved items 2 followup items	Violation: Weld filler material not controlled Violation: Rework to structural beams not controlled Violation: Inadequate inspection of bolting and pipe supports Violation: Lack of inspection status for structural steel Violation: Inadequate plant housekeeping
84-07	5/14 - 5/18/84	1	Large and Small bore pipe supports	None	None

11-4

INSPECTION NUMBER	INSPECTION DATE	NUMBER OF INSPECTORS	AREAS INSPECTED	FINDINGS	VIOLATIONS
84-08	4/30 - 5/25/83	3	NDE van inspection of ASME and structural weldments by independent examination	3 unresolved items	Violation: Film indication not evaluated/ documented Violation: Inadequate review of RT film Violation: Minimum wall violation
84-09	5/14 - 6/15/84	2	Corrective action programs, electrical penetrations, pipe whip restraints, component supports	1 unresolved item 4 followup items	Violation: Adverse trends not detected by analysis Violation: Penetration NDE not performed
84-10	5/21 - 5/24/84	2	Pre-operational security inspection	25 followup items	None
84-11	6/18 - 7/27/84	3	Document control, containment supports, design change control, plant stack, battery installation, equipment preventive maintenance	5 unresolved items 2 followup items	Violation: Hold points violated during pipe whip restraint installation Violation: Inadequate review of design change documents
84-12	cancelled				
84-13	7/30 - 9/6/84	1	Design change control, revetment ditch, pre-op personnel qualifications, hydrotests, preventive maintenance, standby liquid control system, weld filler metal control	4 unresolved items	Violation: Inadequate tread engagement of strainer top bolts Violation: Field issued weld filler metal not controlled
84-14	8/20 - 8/24/84	1	Welder qualifications, welding, welding records	4 unresolved items	None

A-49

INSPECTION NUMBER	INSPECTION DATE	NUMBER OF INSPECTORS	AREAS INSPECTED	FINDINGS	VIOLATIONS
84-15	9/10 - 11/2/84	3	Electrical cable separation, containment penetrations, diesel generator modifications	6 unresolved items 3 followup items	None
84-16	10/29 - 11/2/84	1	Radiological Control staffing	None	None
84-17	10/29 - 11/2/84	1	Safety related equipment installation, inspection of equipment, preventive maintenance	1 unresolved	None

INSPECTION HISTORY

- INITIAL INSPECTION - APRIL, 1972

- MONITORED CONSTRUCTION ACTIVITIES

  SITE PREPARATION AND FOUNDATION
  CONCRETE WORK
  SAFETY-RELATED STRUCTURES
  PIPING AND WELDING
  ELECTRICAL ACTIVITIES
  MECHANICAL COMPONENTS
  INSTRUMENTATION
  QUALITY ASSURANCE PROGRAMS

- TO DATE - 120 INSPECTION REPORTS ISSUED

- COMPARISON OF INSPECTION HOURS (APPROXIMATE)

NINE MILE POINT 2	- 8250
HOPE CREEK	- 7600
SHOREHAM	- 6500
SUSQUEHANNA 1	- 7100
LIMERICK	- 7800

- RESIDENT INSPECTOR LOCATED ON-SITE
  OCTOBER, 1981 - MAY, 1983
  JULY 1983 - PRESENT

ENFORCEMENT RECORD

- NOTICE OF VIOLATION ISSUED FOR ENFORCEMENT
- LICENSEE RESPONDS TO NOTICE WITH PROPOSED CORRECTIVE ACTION
- RESPONSE REVIEWED BY COGNIZANT INSPECTORS AND REGIONAL MANAGEMENT
- CORRECTIVE ACTION IMPLEMENTATION VERIFIED DURING FOLLOWUP ROUTINE INSPECTIONS
- ENFORCEMENT COMPARISON

NINE MILE POINT 2	- 65
SHOREHAM	- 75
SUSQUEHANNA 1	- 102
HOPE CREEK	- 58
LIMERICK	- 86

- ESCALATED ENFORCEMENT ACTIONS

I&E CAT	LEVEL II	100,000 CIVIL PENALTY
FOR QA PROGRAM BREAKDOWN/ STONE AND WEBSTER USE OF TRAINEES TO PERFORM QC INSPECTION	LEVEL III	100,000 CIVIL PENALTY

REGIONAL CONSTRUCTION TEAM INSPECTION

- PERFORMED BY REGION I NOVEMBER-DECEMBER 1981
- COORDINATED TEAM APPROACH

  QUALITY ASSURANCE
  DESIGN CONTROLS
  PROJECT MANAGEMENT
  PROCUREMENT CONTROL
  ELECTRICAL & MECHANICAL CONSTRUCTION
  NONDESTRUCTIVE EXAMINATION

- INSPECTION EFFORT ONSITE - 394 HOURS
- RESULTS: LOW LEVEL VIOLATIONS CITED IN:
  - VIOLATION OF REQUIREMENTS TO ESTABLISH AND EXECUTE AN EFFECTIVE QA PROGRAM INCLUDING A LACK OF NMPC INVOLVEMENT IN AN OVERSIGHT OF QA PROGRAM
  - DESIGN CONTROL IMPLEMENTATION AND REVIEW
  - PROCUREMENT SPECIFICATIONS
- SECOND TEAM INSPECTION - DECEMBER 1984
- PRELIMINARY RESULTS
  - VIOLATION FOR PARTICULATE CONTAMINATION IN PIPING SYSTEMS AND UNDERSIZED WELD ON INSTRUMENTATION SUPPORT.
  - IMPROVED CONDUCT OF QA ACTIVITIES

NDE VAN INSPECTION
APRIL 30 - MAY 25, 1984

- TOTAL OF 662 INSPECTION HOURS

- PURPOSE OF INSPECTION IS TO VERIFY THE ADEQUACY OF SITE NDE PROGRAMS THROUGH INDEPENDENT TESTING

- EXAMINED NUMEROUS WELDS

  35 ASME PIPE WELDS RADIOGRAPHED
  30 ASME PIPE WELDS LIQUID PENETRANT TESTED
  30 WELDS MAGNETIC PARTICLE TESTED
  86 ASME PIPE WELDS VISUALLY EXAMINED
  41 ASME PIPE SUPPORT WELDS VISUALLY EXAMINED
  20 STRUCTURAL WELDS VISUALLY EXAMINED
  34 WELDS ULTRASONICALLY THICKNESS MEASURED
  186 RADIOGRAPHIC FILM RE-INTERPRETED

- THREE VIOLATIONS

  UNACCEPTABLE INDICATION PRESENT ON RT FILM
  MINIMUM WALL THICKNESS VIOLATION
  READER SHEET DID NOT DOCUMENT EVALUATION OF LINEAR INDICATION

- <u>SUMMARY OF NDE STATUS</u>

THE LICENSEE HAS EXAMINED THIRTY NINE (39) ADDITIONAL WELDS TO DETERMINE IF MINIMUM WALL THICKNESS REQUIREMENTS WERE MET. APPROXIMATELY ONE HALF OF THE WELDS WERE FOUND TO BE UNDER MINIMUM WALL THICKNESS REQUIREMENTS. THE LICENSEE DETERMINED THAT THE CONDITION WAS REPORTABLE UNDER 10CFR50.55(e). THE ITEM IS AN OPEN REGION I CONCERN. TO ADDRESS THE TWO OTHER VIOLATIONS IN REGARD TO FILM INTERPRETATION AND ASSOCIATED DOUCMENTATION, THE LICENSEE DIRECTED STONE AND WEBSTER TO REVIEW ALL ITT-GRINNELL RADIOGRAPHS ONSITE FOR CODE ACCEPTABILITY. THE RESULTS HAVE YET TO BE REVIEWED BY REGION I.

SYSTEMATIC ASSESSMENT OF LICENSEE PERFORMANCE
(SALP)

- PURPOSE - TO EVALUATE LICENSEE PERFORMANCE
  - IDENTIFIES AREAS REQUIRING INCREASED/DECREASED REGIONAL INSPECTION RESOURCES
- PREPARED BY REGION I - INPUT FROM
  - RESIDENT INSPECTORS
  - REGIONAL SPECIALIST INSPECTORS
  - NRR PROJECT MANAGER
- NINE MILE POINT 2 SALP RESULTS
  - CYCLE 1 FEBRUARY 1, 1980 - JANUARY 31, 1981
    CATEGORY 2 IN 16 AREAS
  - CYCLE 2 OCTOBER 1, 1981 TO SEPTEMBER 30, 1982
    CATEGORY 1 IN 2 AREAS
    CATEGORY 2 IN 5 AREAS
    CATEGORY 3 IN 1 AREA - QUALITY ASSURANCE
  - CYCLE 3 OCTOBER 1, 1982 TO SEPTEMBER 30, 1983
    CATEGORY 1 IN 1 AREA
    CATEGORY 2 IN 5 AREAS
    CATEGORY 3 IN 2 AREAS -
    PIPING SYSTEMS AND SUPPORTS,
    PROJECT MANAGEMENT/QUALITY ASSURANCE

REGION I OVERVIEW

- PAST INSPECTIONS HAVE DETECTED NUMEROUS DEFICIENCIES
- CURRENTLY LARGE NUMBER (APPROXIMATELY 350) OF OPEN ITEMS FOR WHICH CORRECTIVE ACTIONS REMAIN TO BE VERIFIED, COMPOSED OF
  - 50.55(e) REPORTS, VIOLATIONS, BULLETINS, CIRCULARS, UNRESOLVED, FOLLOWUP ITEMS
- LICENSEE HAS INSTITUTED CORRECTIVE ACTIONS SINCE CAT INSPECTION
  - MANAGEMENT REORGANIZATION
  - HARDWARE REINSPECTIONS
  - PROGRAM ENHANCEMENTS
  - SITE QUALITY TRENDING
- DEFINITIVE STATEMENT REGARDING CONSTRUCTION QUALITY CANNOT BE MADE UNTIL COMPLETION OF VERIFICATION OF CORRECTIVE ACTIONS, ASSURANCE OF INSTALLED HARDWARE DESIGN CONFORMANCE AND SATISFACTORY COMPLETION OF PRE-OPS PROGRAM.

REGION I - NMPC MEETINGS
QA OR PROJECT MANAGEMENT RELATED

JUNE 1, 1982 82-06 ENFORCEMENT CONFERENCE
SAME TOPIC AS 82-08 REORGANIZATION VP-QA AND SR. VP POSITIONS

JULY 16, 1982 82-08 ENFORCEMENT CONFERENCE

ASME C1.1 WELDING PROCEDURES NOT PROPERLY QUALIFIED FOR IMPACT TESTING. ROOT CAUSES FROM NRC STANDPOINT: INEXPERIENCE OF ITT PERSONNEL, INEFFECTIVE SWEC CONTROL OF ITT ACTIVITIES, INADEQUATE NMPC ATTENTION TO CONTRACTOR AND SUB-CONTRACTOR ACTIVITIES. LICENSEE COMMITTED TO: STRENGTHEN ITT QA/ENGINEERING, INCREASE SWEC AUDITS OF ITT, AND IMPOSE SWEC REVIEW OF ITT PLANNERS, INCREASE ITT/SWEC MANAGEMENT INVOLVEMENT

OCTOBER 20, 1982 82-13 ENFORCEMENT CONFERENCE

USE OF SWEC QC TRAINEES TO INSPECT ELECTRICAL INSTALLATIONS. CONTINUED REGION I CONCERN OVER NMPC LEVEL OF INVOLVEMENT WITH PROJECT OVERVIEW AND QA ACTIVITIES OF CONTRACTOR AND SUB-CONTRACTORS. LICENSEE COMMITTED TO INCREASE SITE QA STAFFING AND HAVE ADDITIONAL QA MANAGEMENT INVOLVEMENT

MAY 23, 1983 83-09 MANAGEMENT MEETING

REGION I CONCERN WITH NMPC OVERVIEW OF CONTRACTOR PERFORMANCE IN PARTICULAR ITT QA/QC. LICENSEE COMMITTED TO INCREASED ITT QA STAFFING

AUGUST 30, 1983 83-14 ENFORCEMENT CONFERENCE

ITT VP-QA SUPPRESSED QC PERSONNEL ABILITY TO CONTACT NRC. REGION I EXPRESSED CONTINUED CONCERN WITH REGARDS TO NMPC QA PROGRAM AND CONTROL OF ITT ACTIVITIES.

FEBRUARY 22, 1984 84-03 MANAGEMENT MEETING

DISCUSSED MAC MANAGEMENT REVIEWS OF NMPC (PRE-CAT ORDER), NMPC ORGANIZATION CHANGES, INTERIM SALP/CAT CORRECTIVE ACTION STATUS

NOVEMBER 14, 1984 84-20 MANAGEMENT MEETING

CAT CORRECTIVE ACTIONS, RT F[illegible]M REVIEWS, QPMP, MAC AUDIT (CAT ORDER), HARDWARE VERIFICATION

FEBRUARY 6, 1985 85-05 (IN PREPARATION) MGT MEETING

MAC AUDIT (CAT ORDER), CAT CORRECTIVE ACTIONS, HARDWARE VERIFICATION

I&E CONSTRUCTION APPRAISAL TEAM INSPECTION

- PERFORMED BY I&E HEADQUARTERS NOVEMBER - DECEMBER 1983

- TEAM INSPECTION - AREAS INSPECTED

  ELECTRICAL AND INSTRUMENTATION
  MECHANICAL
  WELDING AND NONDESTRUCTIVE EXAMINATION
  CIVIL AND STRUCTURAL
  MATERIAL TRACEABILITY, STORAGE AND MAINTENANCE
  QUALITY ASSURANCE

- INSPECTION EFFORT ONSITE 1900 HOURS
- RESULTS LEVEL II VIOLATION INDICATIVE OF QA PROGRAM BREAKDOWN
  $100,000 CIVIL PENALTY

  MAJOR DEFICIENCIES WITH PIPE WELD RADIOGRAPHS
  DOCUMENT CONTROL PROGRAM DEFICIENT
  INADEQUATE NIAGARA MOHAWK MANAGEMENT INVOLVEMENT
  INADEQUATE QC INSPECTION PERFORMED
  UNTIMELY IMPLEMENTATION OF CORRECTIVE ACTIONS

MR. CHARLES V. MANGAN

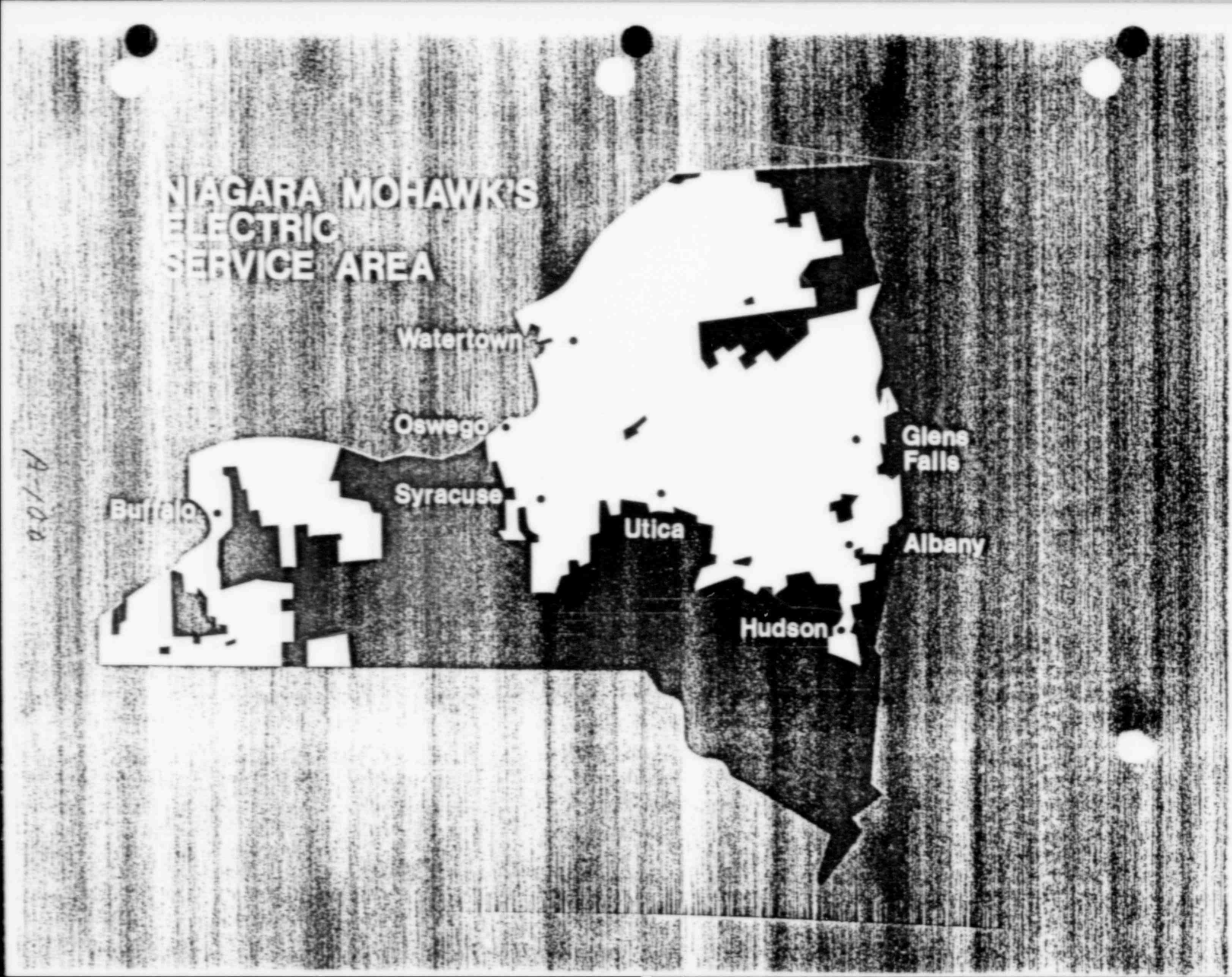
Vice President

Nuclear Engineering & Licensing

NIAGARA MOHAWK

A-99

APPENDIX VI
NIAGARA MOHAWK PRESENTATION -
MANAGEMENT PHILOSOPHY


NIAGARA MOHAWK'S
ELECTRIC
SERVICE AREA
Watertown
Oswego
Syracuse
Utica
Glens
Falls
Albany
Hudson

- Niagara Mohawk has 30 Years of Nuclear Experience

- Nine Mile Point Unit 1 has 15 Years of Operating Experience

A-101

NIAGARA MOHAWK

NUCLEAR EXPERIENCE

1953 - Fermi Unit 1 Design

1958 - Peach Bottom Unit 1

1959 - Nuclear Engineering Section

1960 - Vallecitos Design

1963 - Nine Mile Point Unit 1 Design

1969 - Nine Mile Point Unit 1 Operation

1969 - James A. Fitzpatrick Design/Operation

1974 - Nine Mile Point Unit 2 Construction Permit

NIAGARA MOHAWK

A-102

MR. BILLY G. HOOTEN

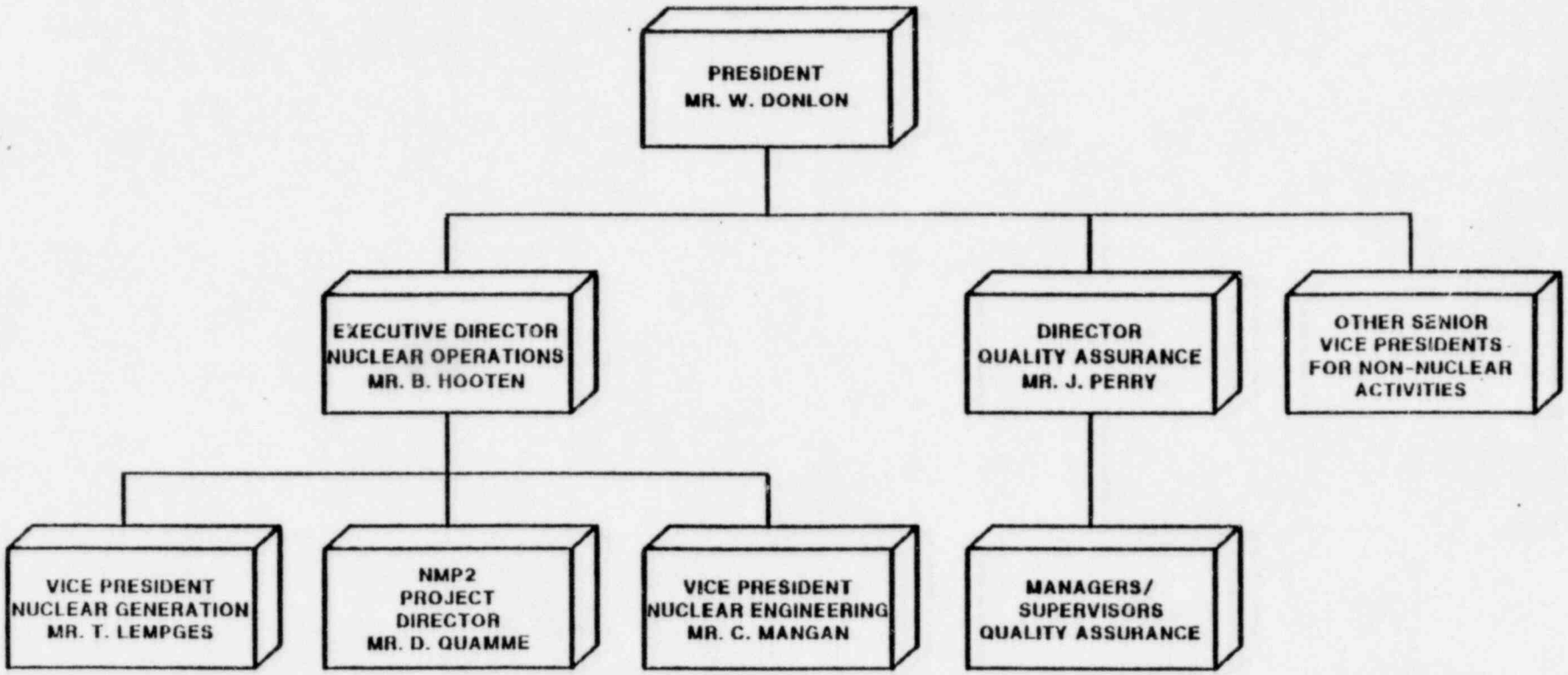
Executive Director
Nuclear Operations

A-103

APPENDIX VII
NIAGARA MOHAWK PRESENTATION -
ORGANIZATION & MANAGEMENT

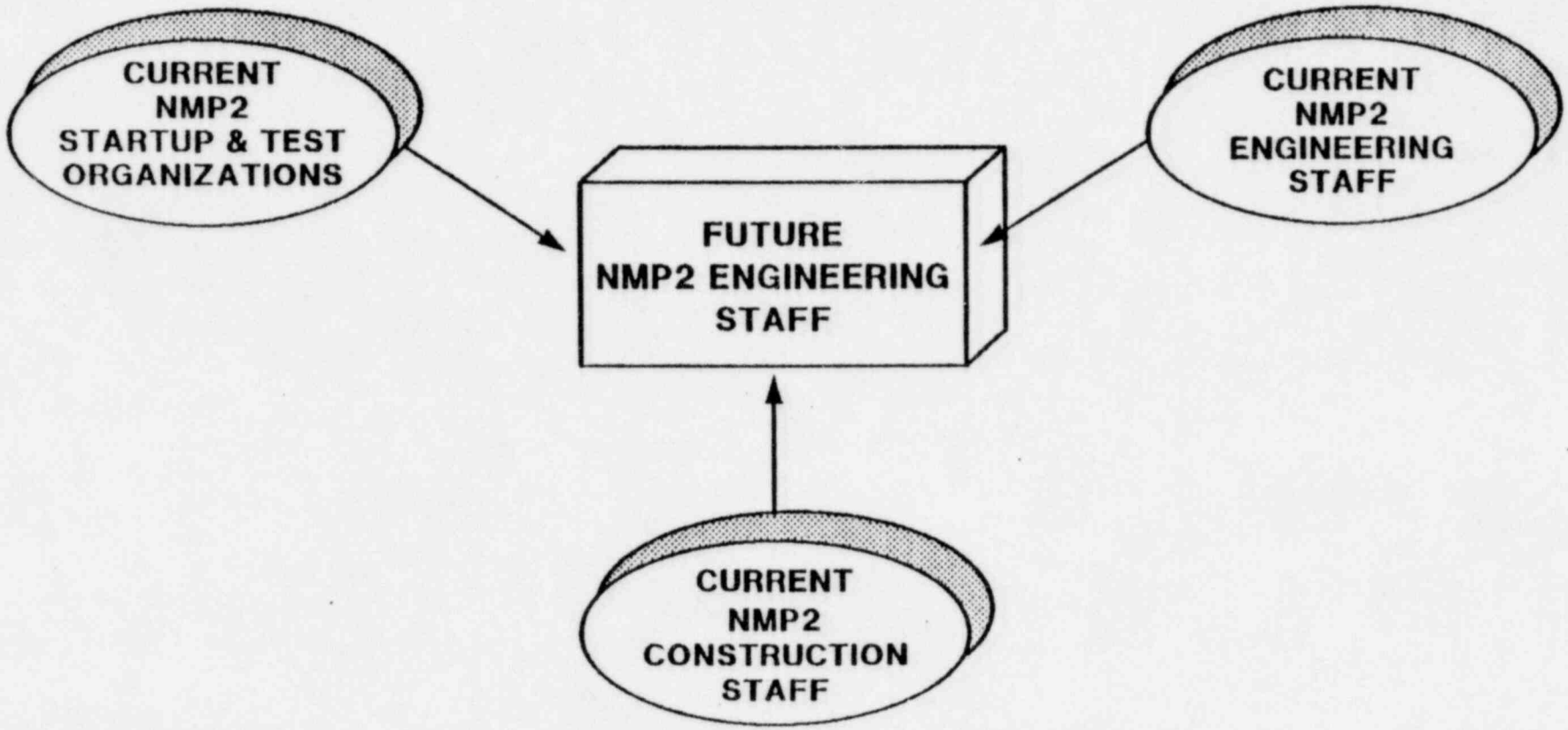
NIAGARA MOHAWK

NMPC UPPER MANAGEMENT
NUCLEAR ORGANIZATION



NIAGARA MOHAWK

A-104



NIAGARA MOHAWK

A-106

JAMES PERRY

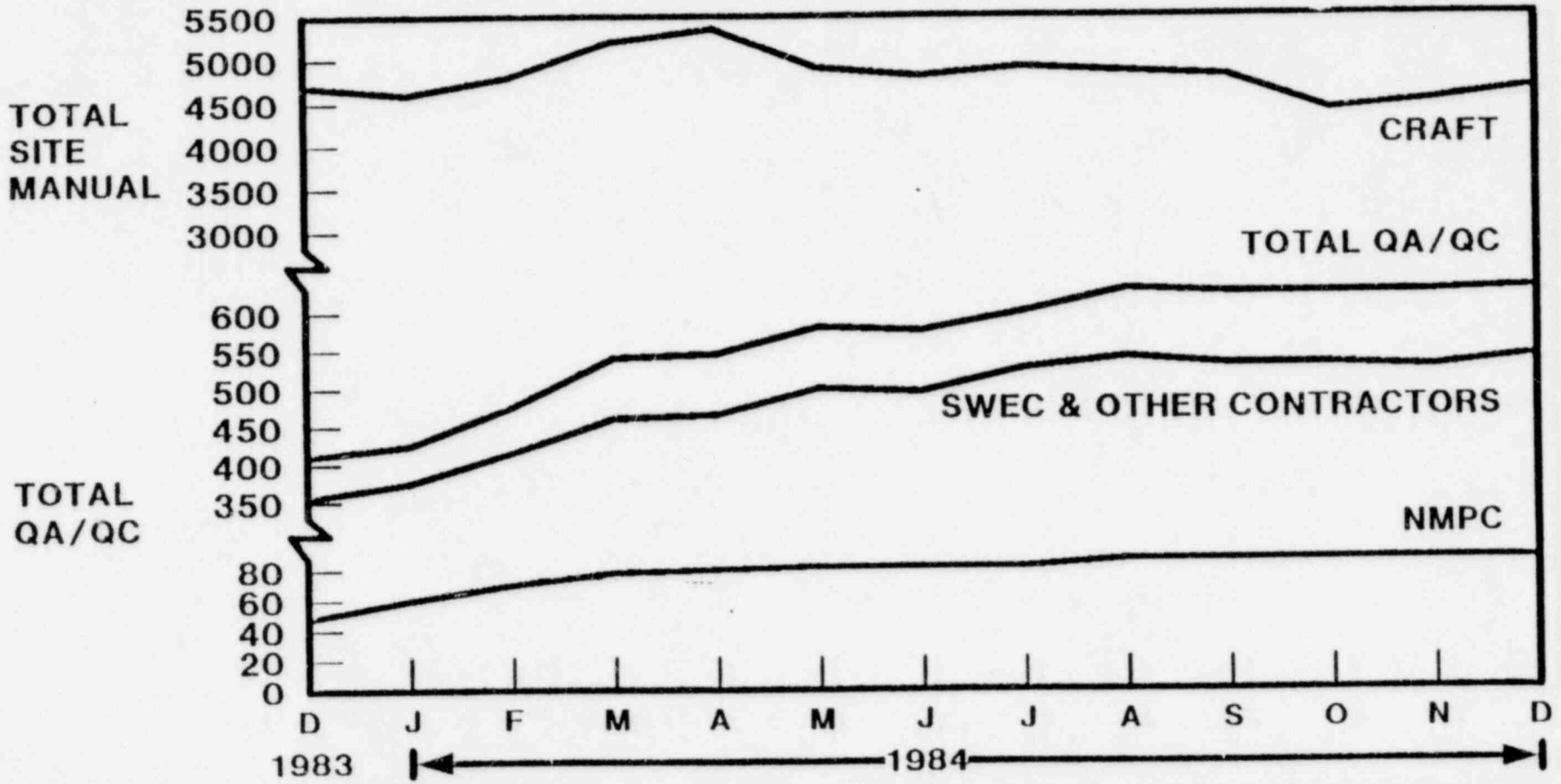
Director - Quality Assurance

NIAGARA MOHAWK

APPENDIX VIII
NIAGARA MOHAWK PRESENTATION - NUCLEAR
QUALITY ASSURANCE ORGANIZATION

NINE MILE POINT UNIT 2 QUALITY ASSURANCE

NMP2 MANPOWER (TOTAL SITE MANUAL VS. TOTAL QA/QC NON-MANUAL)



NIAGARA MOHAWK

A-167

NINE MILE POINT UNIT 2
QUALITY ASSURANCE

A-106

NMPC Current QA Department Staffing and Experience

I. Total Personnel in QAD 194

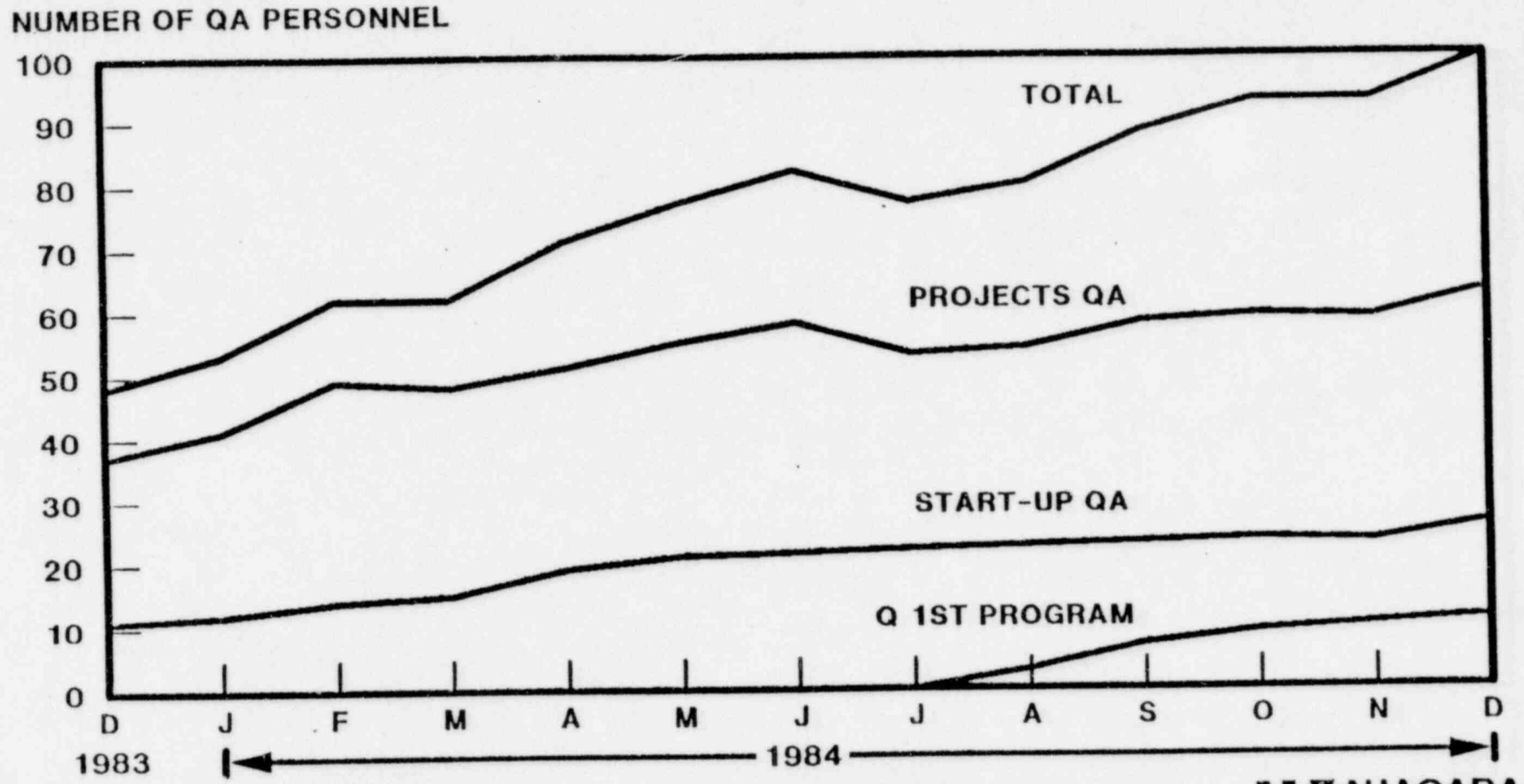
II. Total Individuals with College Degrees 117

III. Total Years of Nuclear Experience 1,221

NIAGARA MOHAWK

NINE MILE POINT UNIT 2 QUALITY ASSURANCE

QA PERSONNEL WITHIN NMPC ORGANIZATION AT NINE MILE POINT UNIT 2 SITE



NIAGARA MOHAWK

A-106

NINE MILE POINT UNIT 2
QUALITY ASSURANCE

NMPC Summary of MAC Independent Assessment Final Report by Project Phase

	Phase	Items Assessed	Unsat. Items	CAR's Issued	IR's Issued	Observed % Acceptable	No. of Recom.
I.	CAT	365	37	12*		89.9	24
II.	SALP	192	1	1		99.5	8
III.	NMPC DEF.	189	4	4		97.9	38
IV.	CONTR. DEF.	2,644	90	62*	6	96.6	150
	Total	3,390	132	77	6	96.1	220

Note: *CAR's No. 42 and 140 Include Phase I & IV and are Shown in Both Areas

NIAGARA MOHAWK

NINE MILE POINT UNIT 2
QUALITY ASSURANCE

QA Program Assessment

- NMPC Conducted Assessment of Contractors' QA Program Effectiveness During August/ September 1984
- Results Indicated Many Program Elements Effective, However, Some Elements Needed Strengthening
  - NMPC Issues CAR's and Specific Recommendations to Enhance QA Programs
- Actions have been Initiated by Contractors and Site Project Management to Increase QA Programs Effectiveness on the Project

NIAGARA MOHAWK

A-111

NINE MILE POINT UNIT 2
QUALITY ASSURANCE

Hardware Surveillance/Inspection Assessment

- As Part of NMPC Surveillance/Inspection Activities, Assessment of Contractor-Accepted, Safety-Related Hardware was Conducted
- Assessment was Performed in Last Quarter of 1984 and included a Re-Inspection of a Sample of Hardware Involving 15 Different Commodities
- Although Some Deficiencies were Identified, Based on Engineering's Evaluation of Deficiencies, have Concluded that Sufficient Confidence Exists to Provide Adequate Assurance that Hardware Installed will Perform its Intended Function

A-112

NIAGARA MOHAWK

NINE MILE POINT UNIT 2 QUALITY ASSURANCE

Nonconformance and Disposition Reports vs. Significant Deficiency Reports

	Number	% Dispositioned Accept As-Is
• N & D's		
- Total N & D's Issued Through November 1984 Cat I, II, and III	9,422	45%
• Significant Deficiency Reports (Ref. Title 10 Part 50.55e Reportable)		
- Total Called into NRC as of November 1984	145	

• $\frac{\text{Total Significant Deficiencies}}{\text{Total N \& D's}} = \frac{145}{9,422} = 1.5\%$

A-113

NIAGARA MOHAWK

NINE MILE POINT UNIT 2 QUALITY ASSURANCE

Allegations/QIP Reported Concerns

- Allegations
  - Allegations Reported to NRC Brought to NMPC's Attention 11 Total
- NMPC Quality First Program Reported Concerns
  - Program Started September 14, 1984. Data Through January 19, 1985

	Quality	
	CAT I	BOP
• Concerns Reported	27	20
• Investigations Complete	20	16
• % of Investigated Concerns Valid	20%	19%

NIAGARA MOHAWK

RICHARD B. ABBOTT

Station Superintendent -
Nine Mile Point 2

A-115

NIAGARA MOHAWK

APPENDIX IX
NIAGARA MOHAWK - PRESENTATION -
OPERATIONS STAFFING & TRAINING

ORGANIZATION AND STAFFING

- Experienced Personnel from Startup and Operation of Fitzpatrick and Nine Mile Point Unit One
- 16 Years of Safe Operation of Nine Mile Point Unit One
- Well Prepared Site Support Organization in Place and Functioning
- Highly Experienced Supervisory Personnel on Shift
- Proven Commitment to Plant Safety

NIAGARA MOHAWK

A-116

ORGANIZATION AND STAFFING

Nine Mile Point 2 Operations Dept. Supervision Qualification Summary

Name	Previous License(s)	Current License	Experience Years: Total Power Plant	Experience Years: Total Nuclear	Experience Years: BWR Commercial	Degree
M. Jones Supv. Ops.	RO NMP1	SRO NMP1	13	13	13	B.S.
R. Gayne Asst. Supv. Ops.	RO NMP1 RO JAF SRO JAF	SRO NMP1	16	16	16	-
W. Wambsgan Asst. Supv. Ops.	RO NMP1 RO JAF SRO JAF	SRO NMP1	16	16	16	-
O. Henderson Supv. Radwaste	-	-	7	7	4	B.S.
M. Bullis Asst. Supv. Radwaste	-	-	3	3	3	-

A-117

NIAGARA MOHAWK

ORGANIZATION AND STAFFING

Nine Mile Point Unit 2 Station Shift Supervisors Qualification Summary

			Experience Years			
Name	Previous License(s)	Current License	Total Power Plant	Total Nuclear	BWR Commercial	Degree
A. Anderson	RO JAF SRO JAF	SRO NMP1	14	14	10	A.A.S.
M. Colomb	RO JAF	SRO NMP1	16.5	14.5	14.5	A.A.S.
W. Davey	RO JAF	SRO NMP1	16.5	16.5	10.5	A.A.S.
A. Degracia	-	RO NMP1	9	9	9	A.A.S./B.A.
J. Kibbe	RO JAF	SRO NMP1	16	16	16	-
G. Moyer	SRO JAF	SRO NMP1	26	26	10	A.A.S.
J. Poindexter	RO NMP1	SRO NMP1	9	9	9	-
D. Topley	-	SRO NMP1	7.5	7.5	7.5	B.S. P.E.
E. Townsend	RO JAF	SRO NMP1	16	16	12	-

NIAGARA MOHAWK

TRAINING

- Non-Licensed Staff Training to Support Nine Mile Point
- Non-Licensed Operator Training Insures Eligibility to become Licensed Operator
- Licensed Operator Training Program Stresses Necessary Knowledge and Skills
- Plant Reference Simulator Available for Cold License Training
- Experienced Training Personnel including SRO and Professional Training Certification

A-115

NIAGARA MOHAWK

TRAINING

Simulator

- Meet ANS/ANSI 3.5 (1981) and Regulatory Guide 1.149
- Features
  - Freeze
  - 20 Initial Conditions
  - Fast Time/Slow Time
  - Snap Shot
  - Back Track
- Updating
  - Once per Year
  - 18 Months after Commercial
- Available for First Cold License Class

A-120

NIAGARA MOHAWK

A-121

CARL D. TERRY

Manager - Nuclear Engineering

NIAGARA MOHAWK

DECAY HEAT REMOVAL

Enlargement of Reactor Building and Primary Containment

- NMP2 Containment Design is an Enhancement in Traditional Mark II Design
- Representative of NMPC Design Philosophy
- Reactor Building Enlarged by Providing North and South Auxiliary Bays
- Relieves Congestion Typical Inside Most Facilities
- Auxiliary Bays Enhance Operability and Reliability of RHR and ECCS Equipment

A-122

NIAGARA MOHAWK

DECAY HEAT REMOVAL

Essential Core Cooling Equipment Location

- North Auxiliary Bay
  - LPCS Pump
  - RHR Pump and Heat Exchanger (Loop A)
- South Auxiliary Bay
  - RHR Pump and Heat Exchanger (Loop B)
  - RHR Pump (Loop C)
- Reactor Building
  - HPCS Pump
  - RCIC Turbine and Pump

A-123

NIAGARA MOHAWK

DECAY HEAT REMOVAL

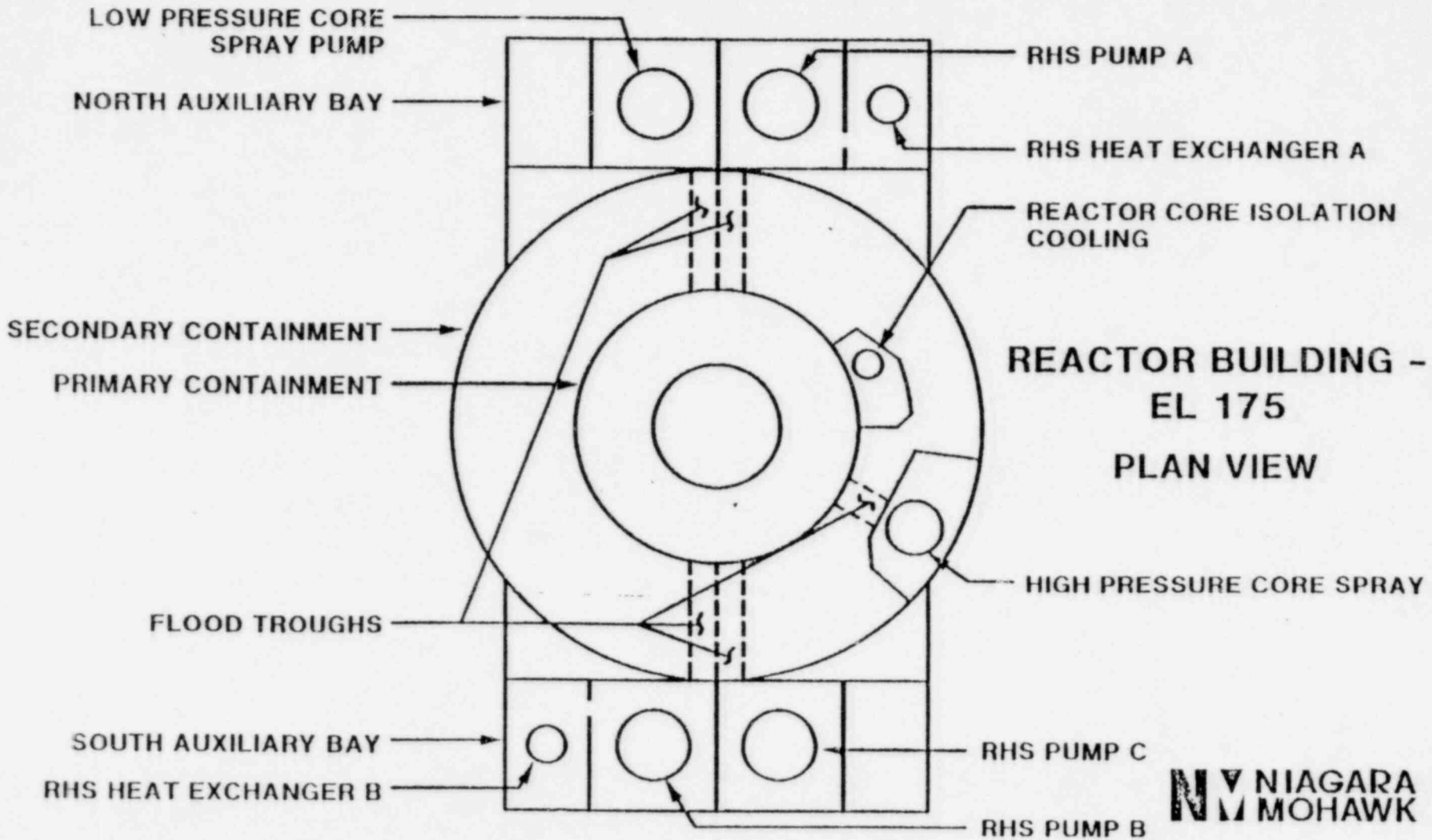
Enhanced Pool and Primary Containment

- Enlarged
- Improved Cleanliness of Water
- Precluded Long Term Pool Degradation

A-124

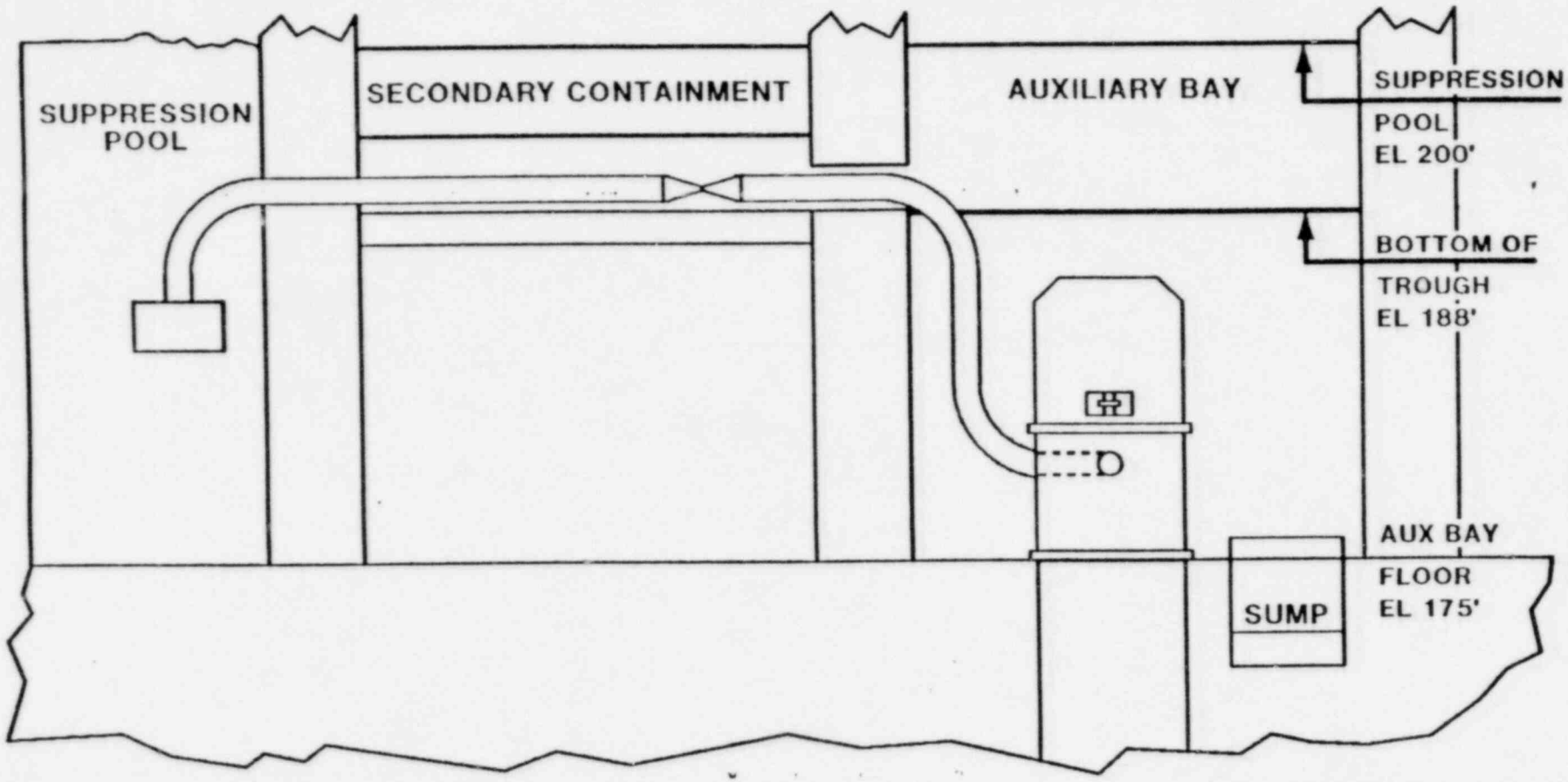
NIAGARA MOHAWK

PHYSICAL SEPARATION CRITERIA FOR DECAY HEAT REMOVAL SYSTEMS
NINE MILE POINT - UNIT 2



NIAGARA MOHAWK

FLOOD TROUGH ELEVATION



NIAGARA MOHAWK

APPENDIX XI
FLOOD THROUGH ELEVATION

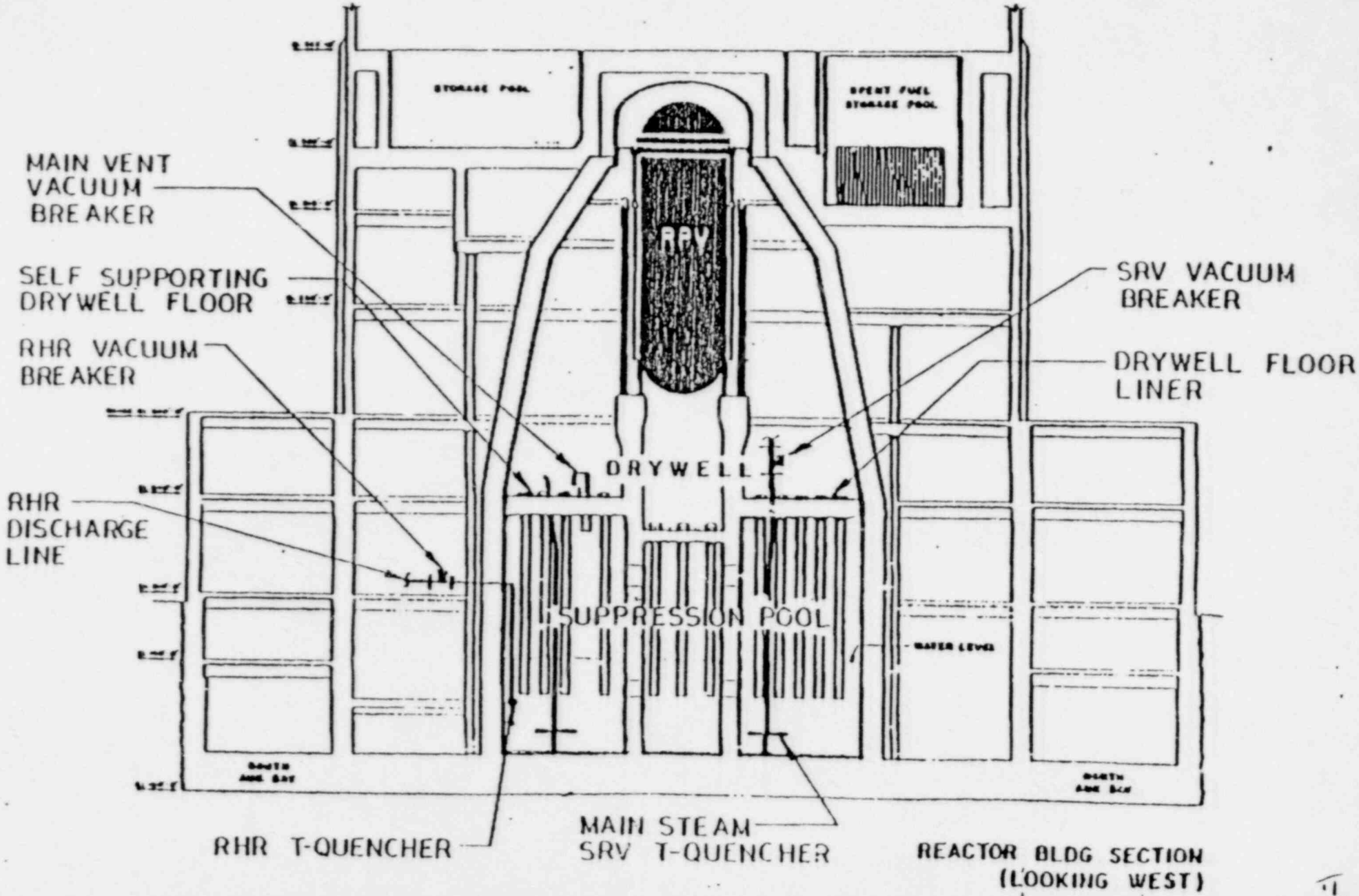
EDWARD R. KLEIN

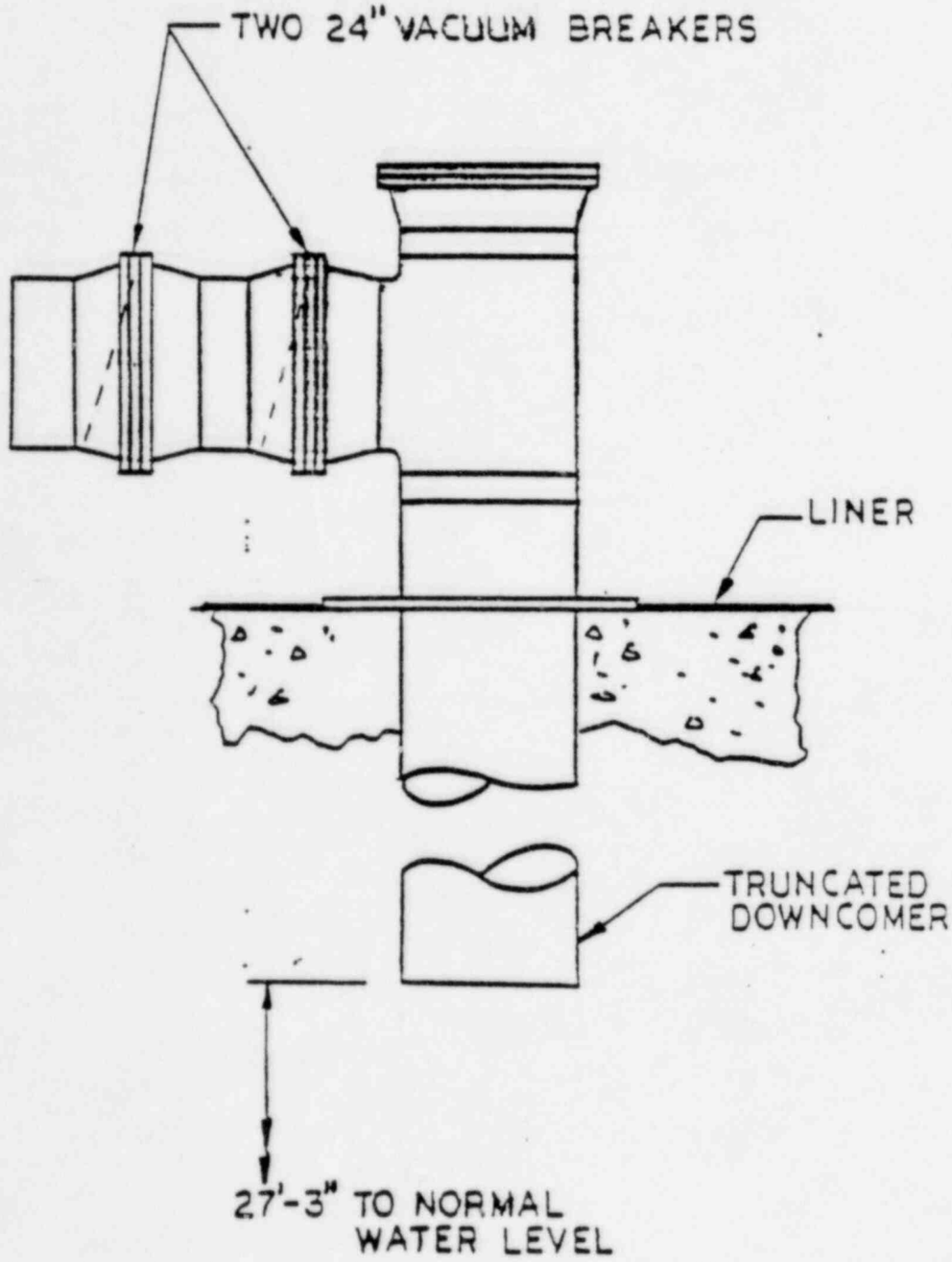
Assistant Manager
Project Engineering

NIAGARA MOHAWK

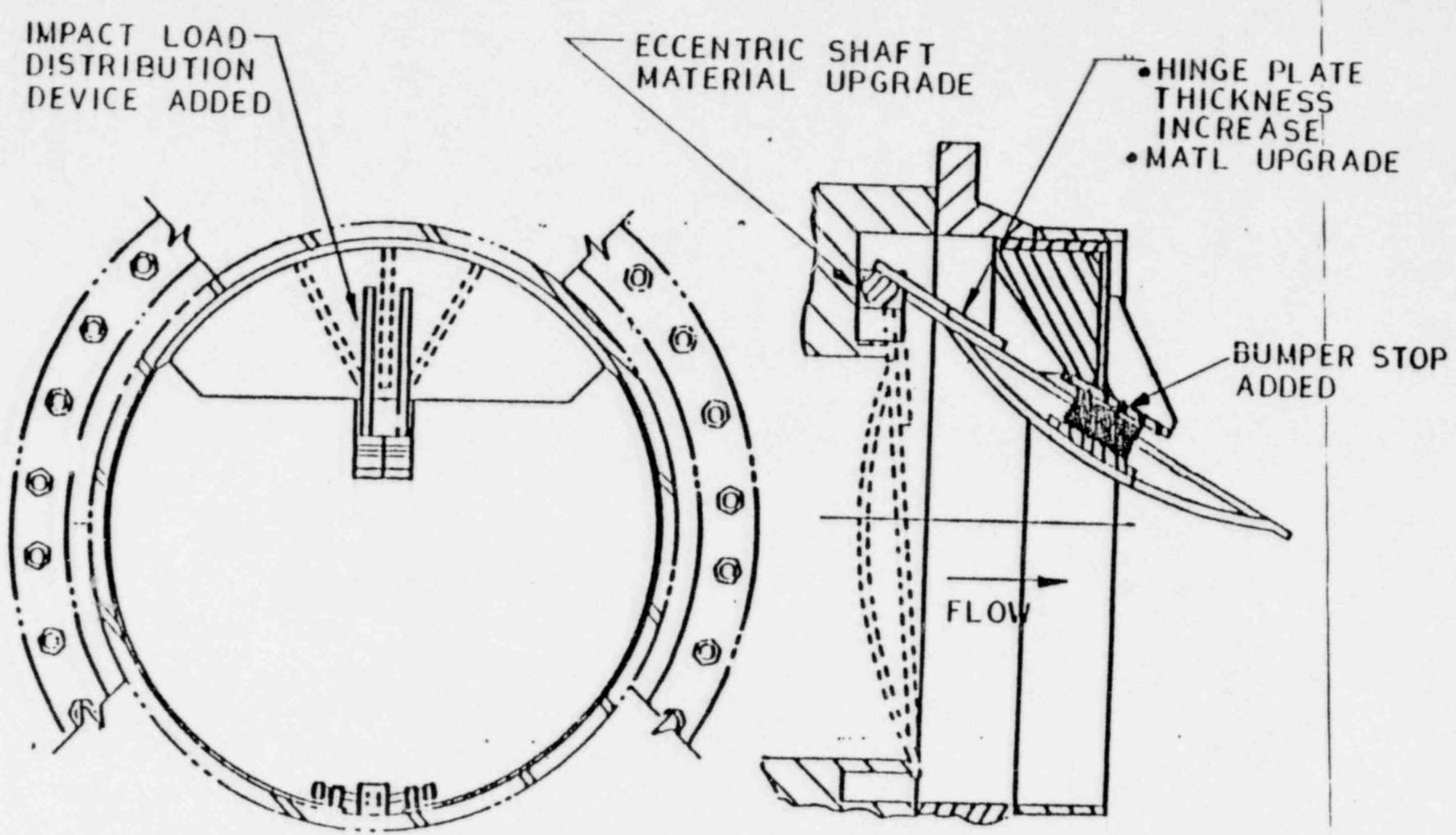
APPENDIX XII
NIAGARA MOHAWK PRESENTATION -
MARK II CONTAINMENT UNIQUE FEATURES

A-127


MAIN VENT VACUUM BREAKER
SELF SUPPORTING DRYWELL FLOOR
RHR VACUUM BREAKER
RHR DISCHARGE LINE
SRV VACUUM BREAKER
DRYWELL FLOOR LINER
RPV
DRYWELL
SUPPRESSION POOL
RHR T-QUENCHER
MAIN STEAM SRV T-QUENCHER
REACTOR BLDG SECTION (LOOKING WEST)



VACUUM BREAKERS



VACUUM BREAKER MODIFICATIONS

NORMAN L. RADEMACHER

Nuclear Design Coordinator
Nine Mile Point Unit 2

A-131

NIAGARA MOHAWK

APPENDIX XIII
NIAGARA MOHAWK PRESENTATION ON ATWS

ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS)

Compliance to 10CFR50.62

- Unit 2 is Installing an Alternate Rod Insertion Subsystem
- Unit 2 is Installing Recirculation Pump Trip Subsystem
- Unit 2 is Installing an Automatically Initiated Standby Liquid Control System Which Provides 86 GPM Equivalent of 13 Weight Percent Boron
- Unit 2 is in Conformance with 10CFR50.62

A-132

NIAGARA MOHAWK

ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS)

SCRAM Reduction

- Operating Personnel Experience
- Improved Equipment and System Design to Reduce SCRAMS
- Enhanced Quality Programs on Nonsafety-Related Equipment

NIAGARA MOHAWK

A-133

ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS)

CRD Hydraulic System Changes

- Materials Upgraded to Low Carbon Stainless Steel to Mitigate IGSCC Concern
- Insert and Withdraw Lines Designed Using "Fast SCRAM" Hydrodynamic Loads
- HCU Pilot SCRAM Solenoid Valves Refurbished

A-124

NIAGARA MOHAWK

ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS)

Improvements in Component/System Design Reduce SCRAMS

- Three 50% Capacity Feedwater Strings
  - One Pump in Standby
  - Alternate Pump "C" Feeds
- Use of Rosemount Analog Trip System
- Recirc. Pump Runback on Partial Loss of Feedwater

A-135

NIAGARA MOHAWK

ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS)

Operating Personnel Experience

- A Prime Contributor to Unnecessary SCRAMS is Inadvertent Actions by Personnel During Surveillance Testing
  - Carefully Evaluated in Preparing Surveillance Test Procedures
  - Experienced Operating Personnel
  - NMP1 Has Not Experienced a SCRAM Caused by Surveillance Test Since 1974
- NMP1 Overall Experience is 5.4 SCRAMS Per Year
  - Industry Average is 6.4/Yr/Plant
  - NMP1 Average Between 1972 and 1984 (Nov.) is 3.25 SCRAMS/Yr
  - NMP1 Average for Past Five Years is 1.4 SCRAMS/Yr

A-136

NIAGARA MOHAWK

NORMAN L. RADEMACHER

Nuclear Design Coordinator

Nine Mile Point Unit 2

A-137

NIAGARA MOHAWK

APPENDIX XIV
NIAGARA MOHAWK PRESENTATION
FIRE PROTECTION

ROBERT RAYMOND

Supervisor
Fire Protection
Nuclear

A-138

NIAGARA MOHAWK

FIRE PROTECTION

- Fire Protection Program
- Defense in Depth
- Responsibility of Fire Protection Program
- Personnel Qualifications

NIAGARA MOHAWK

FIRE PROTECTION

Administration Controls

- Fire Brigade
- Quality Assurance

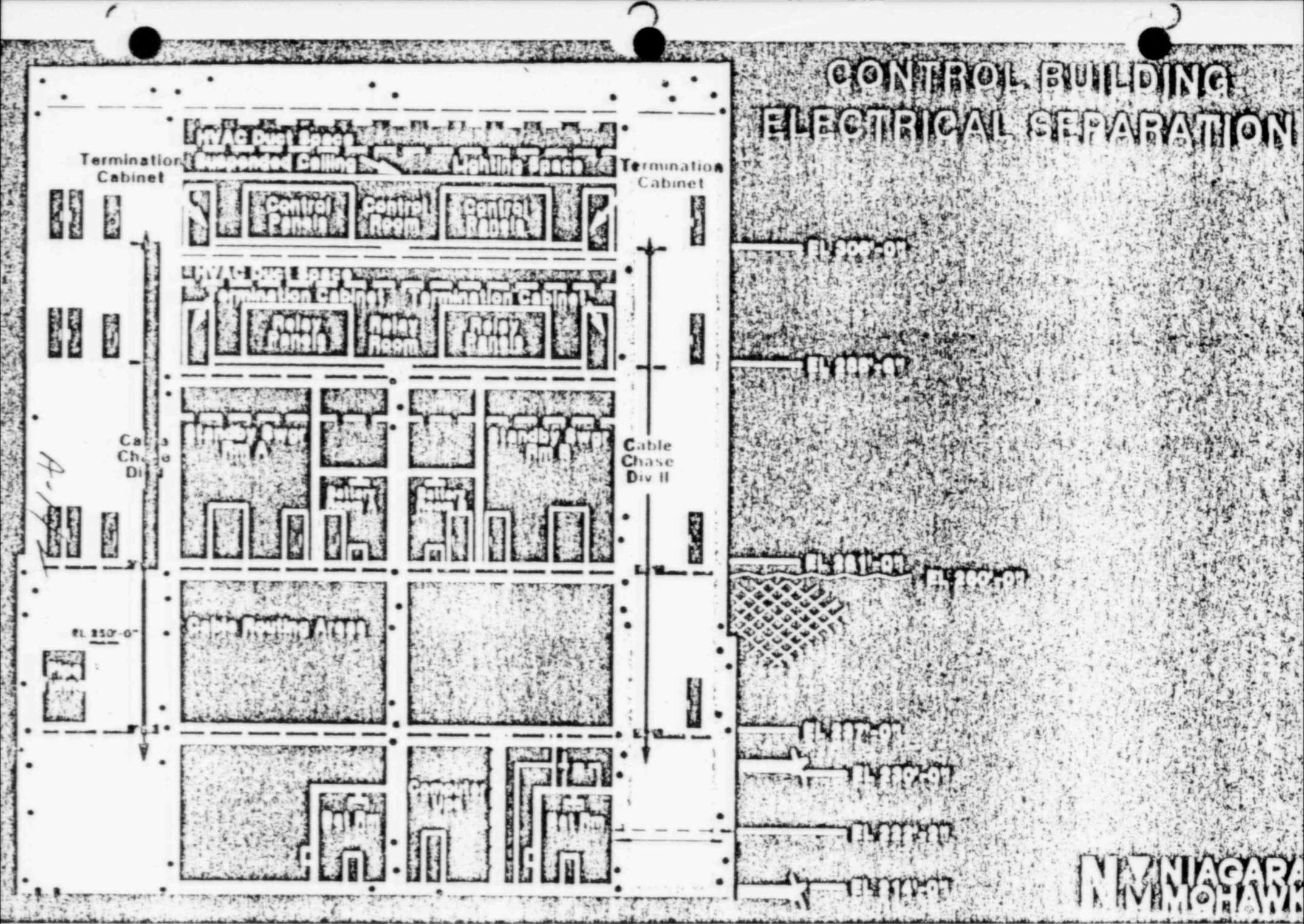
NIAGARA MOHAWK

FIRE PROTECTION

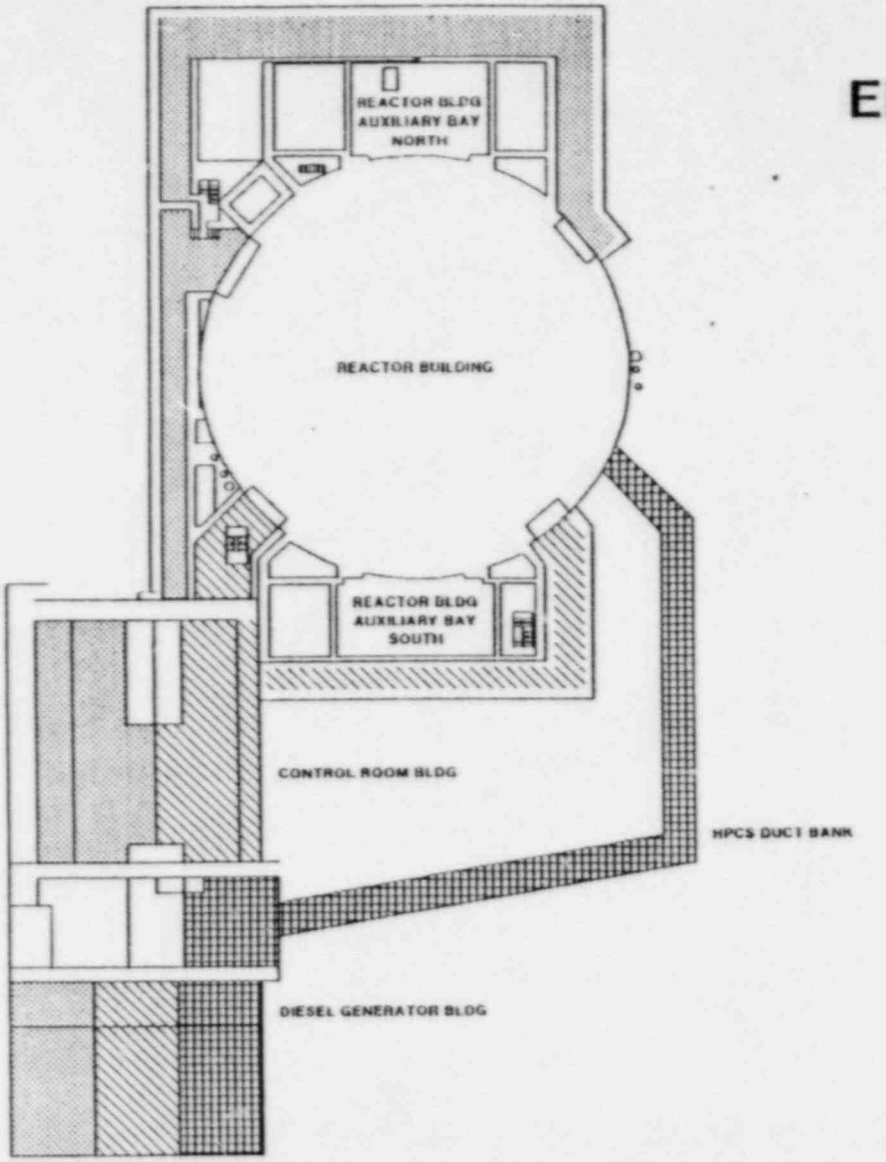
Fire Hazards Analysis

- Fire Hazards/Loading
- Safe Shutdown Analysis
- Associated Circuits

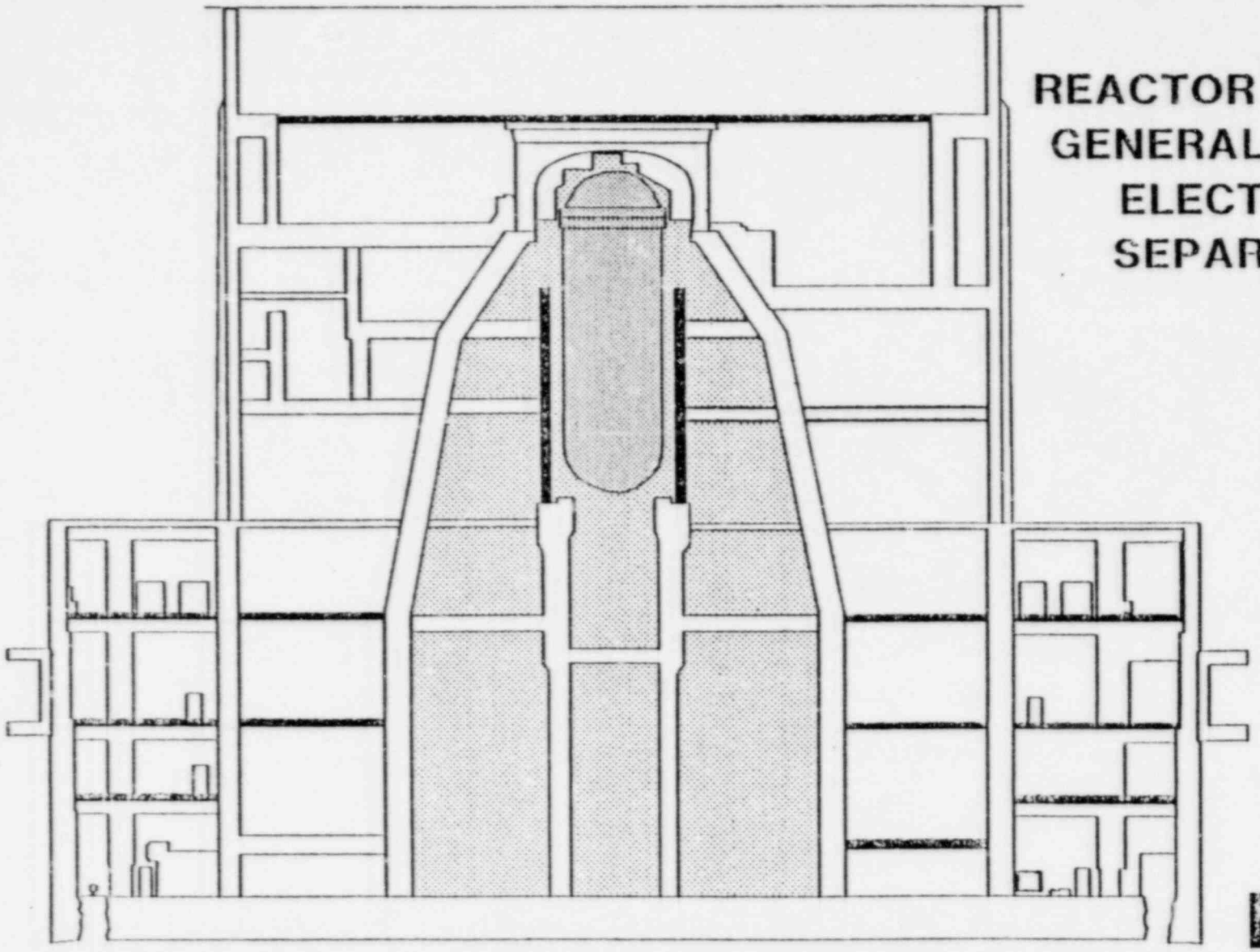
NIAGARA MOHAWK


CONTROL BUILDING
ELECTRICAL SEPARATION
Termination Cabinet
Termination Cabinet
Control Panels
Control Room
Control Panels
Relay Panels
Relay Room
Relay Panels
Cable Chase Div II
EL 250'-0"
EL 300'-0"
EL 290'-0"
NIAGARA MOHAWK

ELECTRICAL SEPARATION



NIAGARA MOHAWK



REACTOR BUILDING GENERAL LAYOUT ELECTRICAL SEPARATION

NIAGARA MOHAWK

A-144

A-145

DOUGLAS PIKE

Asst. Manager, Project Engineering
Systems Design

NIAGARA MOHAWK

APPENDIX XV
NIAGARA MOHAWK PRESENTATION - SPDS
& REMOTE SHUTDOWN CAPABILITY

SAFETY PARAMETER DISPLAY SYSTEM

A-196

NRC Requirements - NUREG 0737

- Concise Display of Critical Plant Variables
- Aid Operator in Determining Safety Status of Plant
- Located in Main Control Room

NIAGARA MOHAWK

SAFETY PARAMETER DISPLAY SYSTEM

Unit 2 Design Features

- Utilize Quadrex/Honeywell System
- Integrated with Radwaste Control & Generator Temperature Monitoring
- Unavailability Goal of .01 for Reactor Operating
- Unavailability Goal of .2 when Shutdown
- Two Redundant 4500 Central Processors with Auto Failover
- UPS Power Supply

NIAGARA MOHAWK

A-147

SAFETY PARAMETER DISPLAY SYSTEM

Displays

- Utilize All But One of BWR Owners Group Displays
- Data Base is Subset of R.G. 1.97 Parameters

NIAGARA MOHAWK

SAFETY PARAMETER DISPLAY SYSTEM

Level I Provides Overall Status

- Real Time Values
- Green/Red Color Coding
- Rate/Trend Information
- Level II Status
- Mode Switch Position

Level II Displays for:

- Reactivity Control
- Core Cooling
- Coolant System Integrity
- Containment Integrity

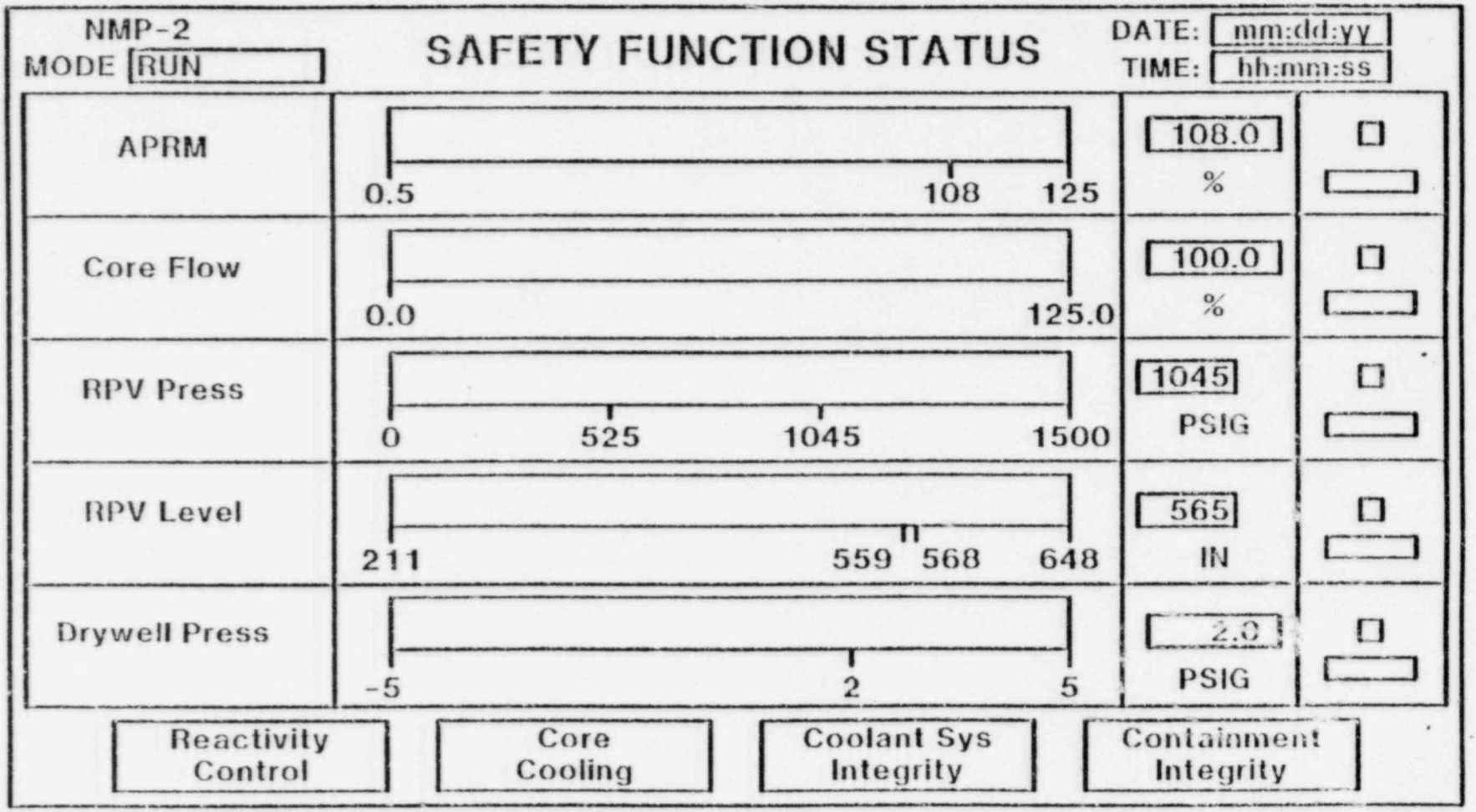
Level II Displays Provide:

- Time History of Past 6 Minutes
- Real Time Values
- Trend/Rate Information
- Status of Level II
- Time, Date, Title
- Mode Switch Position

A-145

NIAGARA MOHAWK

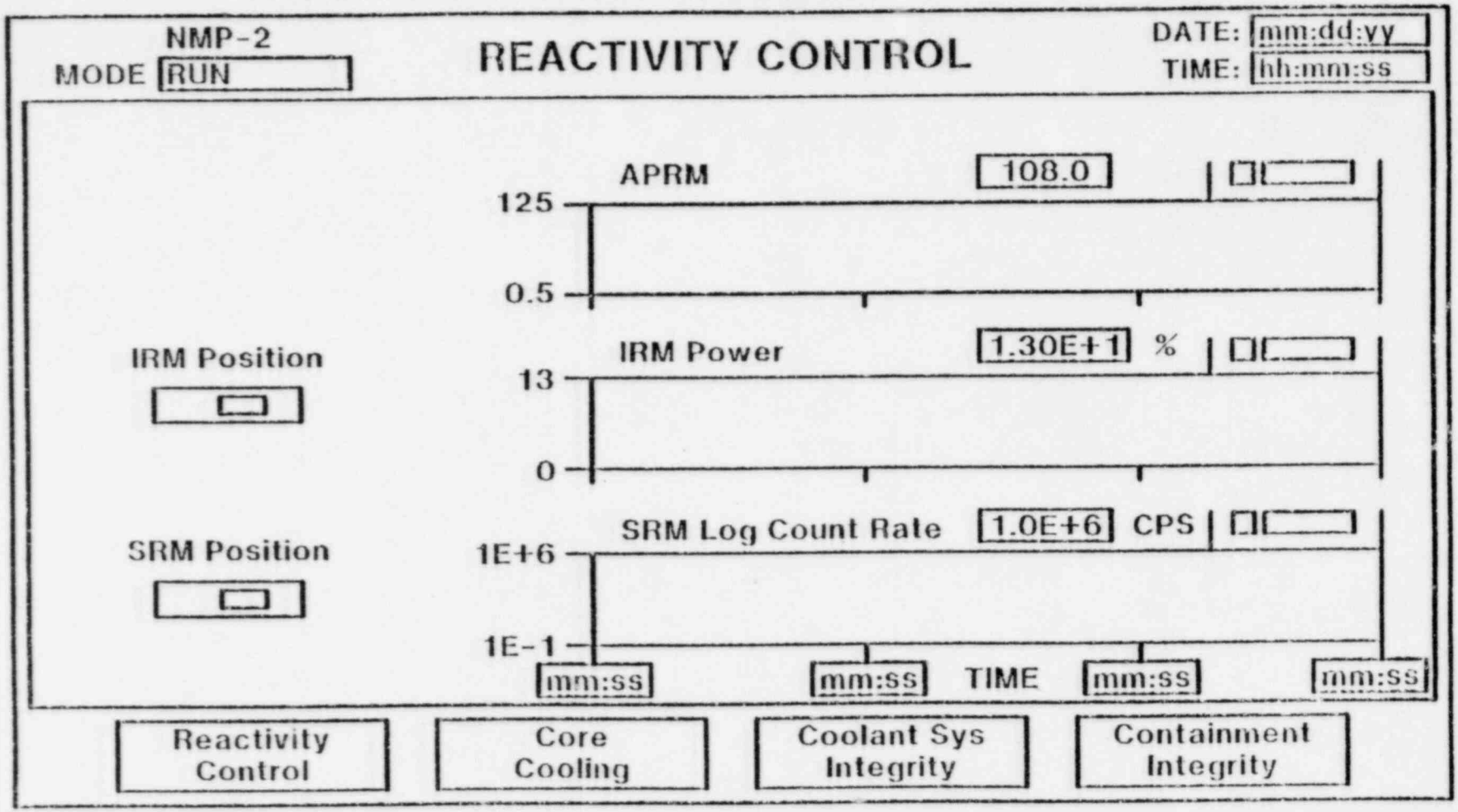
SAFETY PARAMETER DISPLAY SYSTEM



Safety Function Status Display

NIAGARA MOHAWK

SAFETY PARAMETER DISPLAY SYSTEM



Reactivity Control Display

NIAGARA MOHAWK

A-151

SAFETY PARAMETER DISPLAY SYSTEM

Emergency Response Facility System

- R.G. 1.97 Data Base of 552 Analog & Digital Points
- Functions Available in Control Room, TSC, & EOF
  - Graphic Displays (180)
  - Bar Charts
  - Trending of Variables (X-Y Plots)
  - Alarm Summary
  - Group Logs

A-152

NY NIAGARA MOHAWK

SAFETY PARAMETER DISPLAY SYSTEM

ERF Historical Features

- Historical Event Retrieval
  - Trigger Signals
  - 2 Hours Pre-Event/12 Hours Post-Event
  - Recorded in 1, 5 & 30 Second Groups
  - 15 Minute Periods for Up to 14 Days Following 12 Hr. Period
- Historical & Event Trigger Data Accessible from TSC & EOF

A-153

NIAGARA MOHAWK

REMOTE SHUTDOWN CAPABILITY

Nine Mile Point #2 Design

- Remote Shutdown Panels Provided to Meet Requirements
- Remote Panels Fulfills All Appendix R Requirements When Coupled with Some Local Operations

NIAGARA MOHAWK

REMOTE SHUTDOWN CAPABILITY

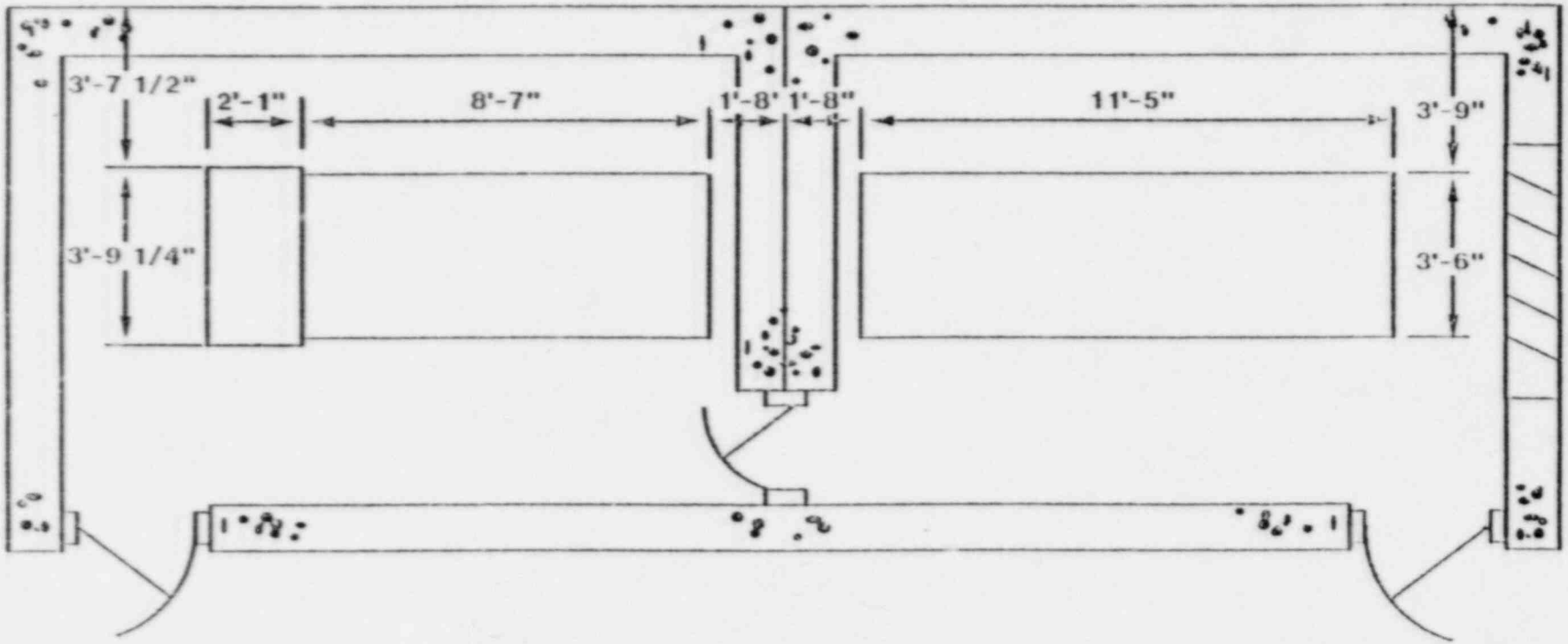
Panel Design Features

- Designed to Same Standards as Main Control Room
- Seismically Qualified
- Class 1E Redundant Instruments and Controls
- Divisional Separation per Regulatory Guide 1.75
- Redundant Controls Separated by Three Hour Fire Barriers
- Positive Pressure Maintained in Rooms

NIAGARA MOHAWK

REMOTE SHUTDOWN CAPABILITY

Remote Control Room



NIAGARA MOHAWK

REMOTE SHUTDOWN CAPABILITY

Physical Independence

- Two Elevations Below Main Control Room
- Electrically Separated by Keylocked Transfer Switches
- Two Safety Related HVAC Systems Physically Separated from Main Control Room HVAC System

NY NIAGARA MOHAWK

REMOTE SHUTDOWN CAPABILITY

Panel Layout and Systems Provided

- Service Water
- Reactor Core Isolation Cooling
- RHR-Shutdown and Suppression Pool Cooling
- SRVs - Four ADS Valves
- Remote Shutdown Room HVAC

A-154

NIAGARA MOHAWK

REMOTE SHUTDOWN CAPABILITY

Panel Layout and Systems Provided - Indication

- Totally Independent From the Control Room
- System Indication for Operability
- Critical Redundant Indication for:
  - RHR Heat Exchanger In/Out Temperature
  - Reactor Shell Flange and Bottom Head Temperature
  - Reactor Level
  - Reactor Pressure
  - Suppression Pool Water Level
  - Suppression Pool Temperature

A-159

NY NIAGARA MOHAWK

REMOTE SHUTDOWN CAPABILITY

Additional Features

- Plant Computer CRT/Keyboard
- Plant Wide Communications System
- Lighting Provided for All Conditions
- Panel will be Included in CRDR for Human Factors Input

A-160

NIAGARA MOHAWK

TWS
ltr

NINE MILE POINT UNIT 2
CHALLENGES TO SAFETY SYSTEMS

A RELIABILITY ANALYSIS* FOR ATWS 3A WAS PERFORMED. THE RESULTS ARE:

- INADVERTENT RRCS SIGNALS RESULTING FROM RANDOM MODE SENSOR AND LOGIC FAILURES IS LESS THAN 1×10^{-4}/YEAR

- INADVERTENT FEEDWATER RUNBACK SIGNALS RESULTING FROM RANDOM MODE FAILURES IN SENSOR AND LOGIC FAILURES IS LESS THAN 1×10^{-4}/YEAR

- IMPACT ON OTHER SAFETY SYSTEMS RESULTING FROM 3A MODIFICATIONS IS NOT SIGNIFICANT

* NEDE 22157

NORMAN L. RADEMACHER

Nuclear Design Coordinator
Nine Mile Point Unit 2

A-162

NIAGARA MOHAWK

APPENDIX XVII
NIAGARA MOHAWK PRESENTATION - PRA
ANALYSIS

PROBABILISTIC RISK ASSESSMENT (PRA)

A-163

Introduction

- A PRA is an Analysis for Core Melt Accidents
- Identifies Sequences which Contribute to Risk
- NMP2 PRA was Performed to Meet NRC Interim Rule in Federal Register (45CFR40101)
- PRA's may be Categorized as Full PRA or Mini PRA

NIAGARA MOHAWK

PROBABILISTIC RISK ASSESSMENT (PRA)

Nine Mile Point Unit 2 Mini PRA

- Identifies Accident Sequences Important to Core Melt Risk
- Includes Loss of Offsite Power as an Accident Initiator
- Does Not Include Other External Events such as Flooding, Fire, Seismic
- Includes Component Level Analysis of Plant Systems
- Includes Various Containment Failure Modes
- Includes Results of Risks, Socioeconomic Impacts

A-164

NIAGARA MOHAWK

PROBABILISTIC RISK ASSESSMENT (PRA)

PRA Inputs

- Accident Event Trees and Functional Success Criteria
- Site Specific Offsite Power Grid Reliability
- Generic Component Failure Rate Data
- One Year of Hourly Meteorological Site Data
- Plant Specific Midlife Population Data
- Plant Specific Topographical Data and Socio-economic Data
- Plant Specific Emergency Planning Information

A-165

NIAGARA MOHAWK

PROBABILISTIC RISK ASSESSMENT (PRA)

PRA Review

- Nuclear Engineering - 2 Nuclear Engineers
- Station Operations - Station Superintendent/SSS
- Project Engineering Manager - Lead Engineers (Structural, Electrical, Mechanical, I&C)
- Environmental Department - Manager and Environmental Engineer
- Radiation Management/Radio Chemistry - Superintendent

A-164

NIAGARA MOHAWK

PROBABILISTIC RISK ASSESSMENT (PRA)

Summary of Nine Mile Point Unit 2 PRA

- Fulfills NRC Requirements for Class 9 Accidents
- Provides an Estimate of Plant Risk
- Provides a Check on Emergency Effectiveness
- Provides an Indication of Risk Mitigating Features
- Provides an Indication of Dominant Risk Contributors

A-162

NIAGARA MOHAWK

PROBABILISTIC RISK ASSESSMENT (PRA)

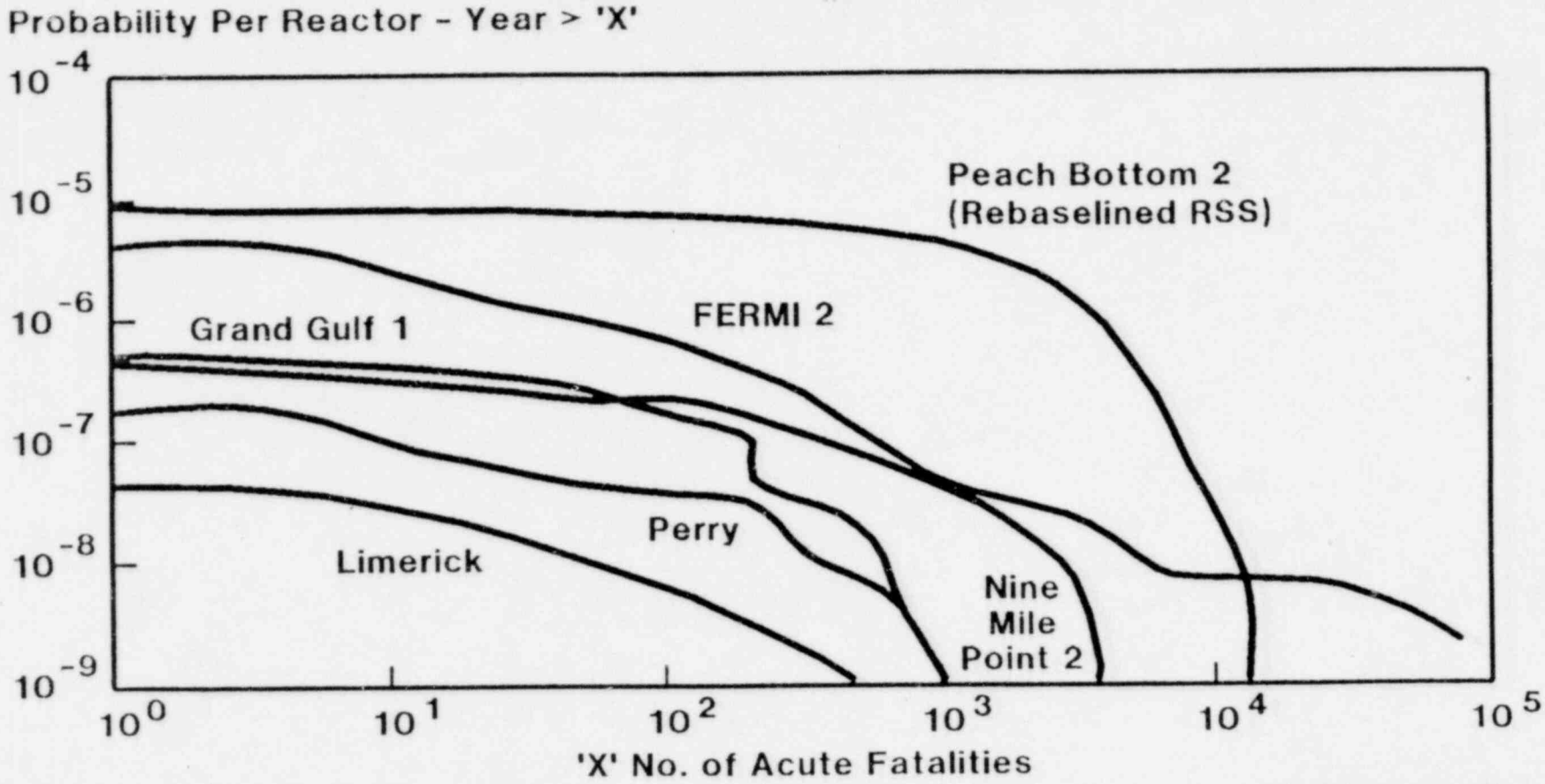
PRA Conclusion

- Further PRA's Include: Unit 1
  Unit 2 (Upgrade)
- Training on the Uses of PRA to Corporate Site Staff
  - Mitigating Core Damage - Operations
  - Engineering Training

NIAGARA MOHAWK

PROBABALISTIC RISK ASSESSMENT (PRA)

Acute Fatalities - BWR Comparison



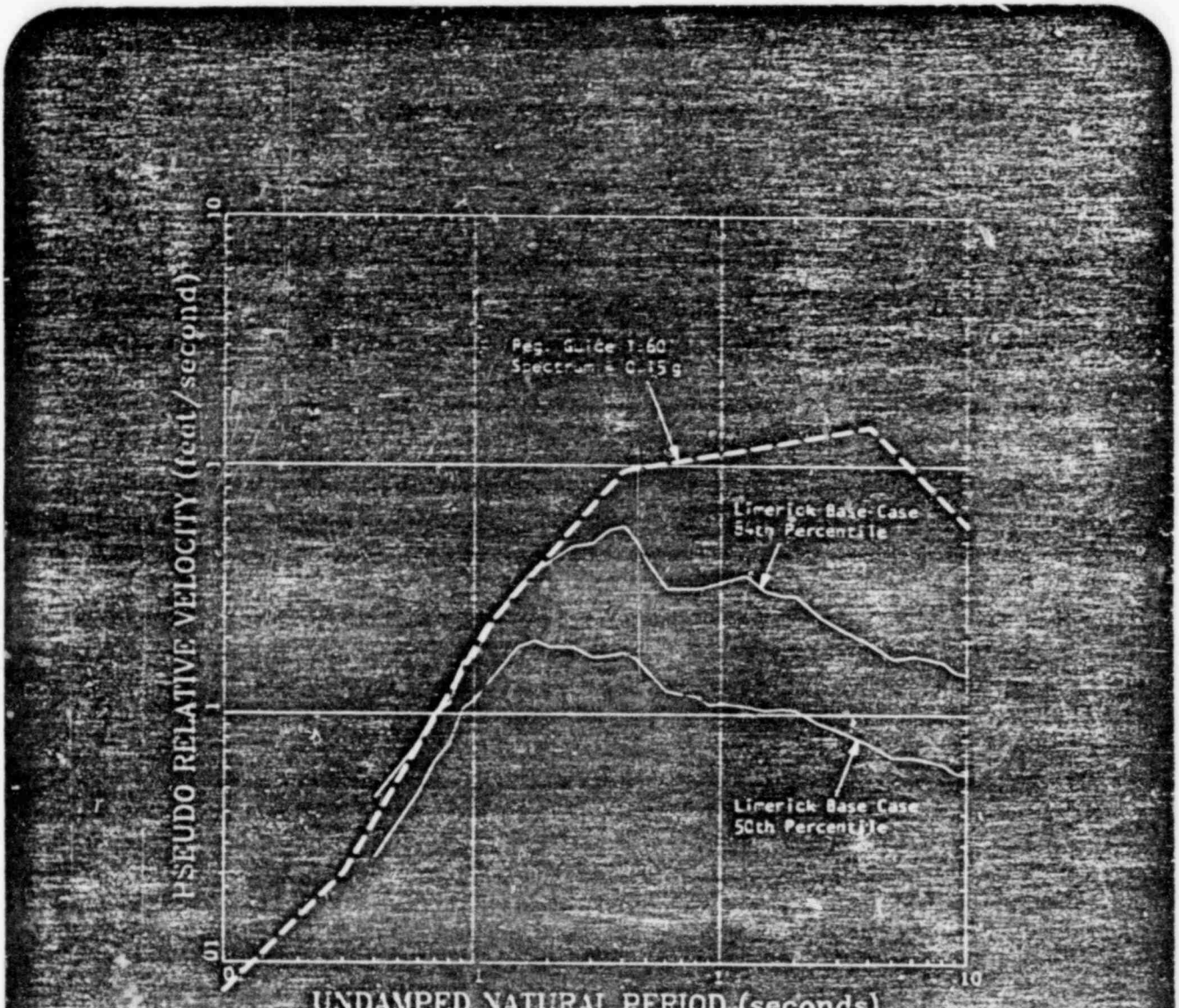
The Y-Axis Units for the Limerick Curves are:
Frequency (Events/Year) > 'X'

NIAGARA MOHAWK

A-16

TABLE 1

Earthquake	Date	Time	Recording Station	Magnitude
Oroville	08/08/75	[illegible]	Station 6	4.9
Oroville	09/27/75		Station 8	4.6
Oroville	09/27/75		Station 9	4.6
Parkfield	06/27/66		Temblor	5.5
Lytle Creek	09/12/70		Allen Ranch	5.4
Friuli	09/11/76	[illegible]	S. Roco	5.5
Cape Mendocino	06/07/75		C. Mendocino	5.3
Helena	10/31/35		Carroll College	5.7
Helena	11/28/85		Federal Building	5.0
Oroville	08/01/75		Seismograph Station	5.7



SITE SPECIFIC SPECTRA
COMPARED TO REG. GUIDE 1.60
FIGURE 6

Agenda for ACRS Committee
Meeting on March 8, 1984
11:00 a.m.
Room 1046, H Street

RECENT SIGNIFICANT EVENTS

Date	Plant	Event	Response	Presented by	Page
ECCS PUMP PROBLEMS					
1/16/85	Hatch 1	HPCI, RCIC Inoperability	Under Study	NRR(Rivenbark)	2
1/31/85	Browns Ferry 1 & 2	RHR Pump Valve Failure	Under Study	NRR(Long)	9
2/7/85	Catawba 1	Both SI Pumps Inoperable	Under Study	IE(Weiss)	15
MISCELLANEOUS PROBLEMS					
1/12/85	Sequoyah 2	Reactor Protection Sys Malfunction	Info Notice IN 85-18 Issued 3/7/85	IE(Weiss)	17
1/28/85	Farley 2	Failure of Containment Tendons	Info Notice IN 85-10 Issued 2/6/85	NRR(Lear)	25

NRR STAFF PRESENTATION TO THE ACRS

SUBJECT: HATCH 1 HPCI, RCIC INOPERABILITY (1/16/85)

DATE: 3/5/85

PRESENTER: GEORGE RIVENBARK

PRESENTER'S TITLE/BRANCH/DIV: PROJECT MGR/ORB#4/DOL

PRESENTER'S NRC TEL. NO.: 49-27136

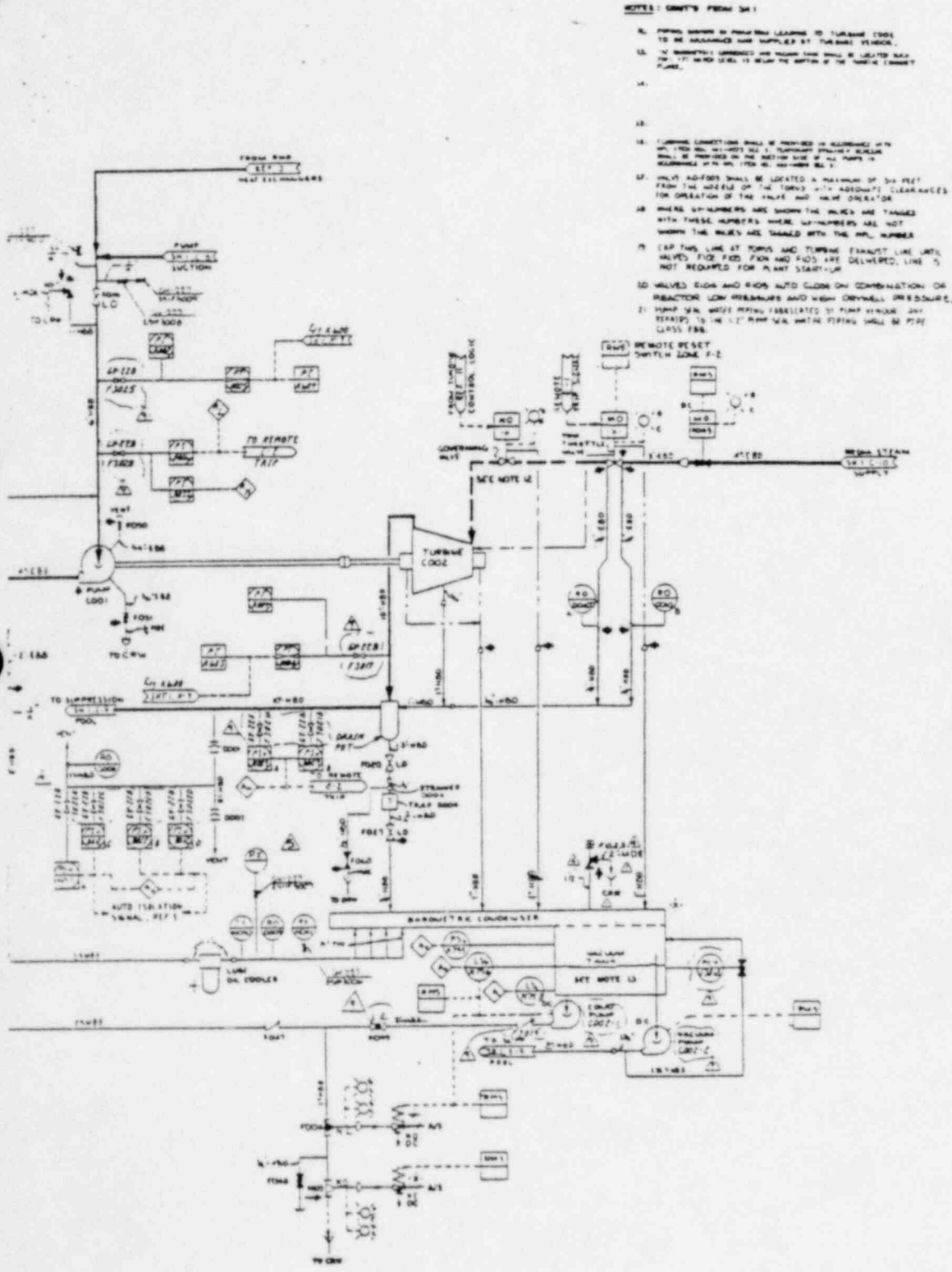
SUBCOMMITTEE: REACTOR OPERATIONS

A-173

2

- HATCH - 1 - HPCI, RCIC INOPERABLE FOLLOWING LOW VESSEL CONDITION JANUARY 16, 1985 (G. RIVENARK)

  - VITAL AC BUS LOST WITH UNIT AT 60% POWER
  - FEEDWATER PUMPS RUNBACK
  - REACTOR TRIP ON LOW VESSEL LEVEL
  - HPCI AND RCIC ACTUATED AUTOMATICALLY ON LOW-LOW VESSEL LEVEL
  - VESSEL LEVEL RESTORED; HPCI, RCIC AND FEEDWATER TURBINES TRIP ON HIGH VESSEL LEVEL
  - VITAL AC BUS RESTORED, FEEDWATER PUMP "A" RESTARTED MANUALLY
  - UNABLE TO RESET RCIC TURBINE TRIP VALVE DUE TO TRIP LINKAGE PROBLEM
  - HPCI TURBINE STOP VALVE STUCK IN HALF OPEN POSITION
  - HPCI AND RCIC SYSTEMS REPAIRED AND TESTED PRIOR TO RESTART OF PLANT



4

H-16336 RE

Georgia Power

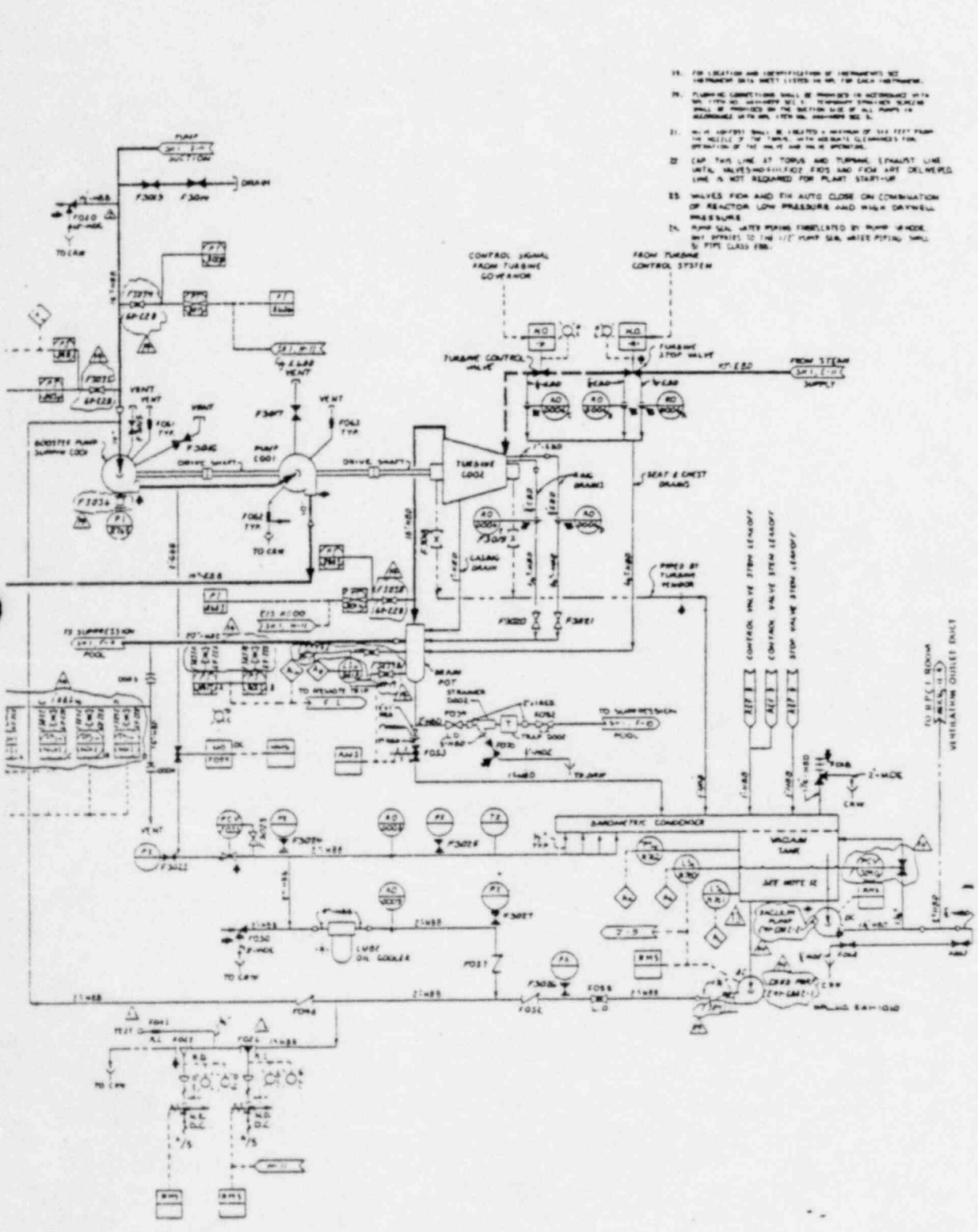
EDWIN I. HATCH
NUCLEAR PLANT – UNIT 1

RCIC SYSTEM
PIPING AND INSTRUMENTAION DIAGR

FIGURE 4.7-1 (SHEET 2 OF 2)

A-175

REV 1



REV 2 7
REV 1 7
H-16333 RE

Georgia Power

EDWIN I. HATCH
NUCLEAR PLANT – UNIT 1

HIGH-PRESSURE COOLANT INJECTION SYSTEM (P&ID)

FIGURE 7.4-2 (SHEET 2 OF 2)

NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES 8/31/85

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
EDWIN I. HATCH, UNIT 1	0 5 0 0 0 3 2 1	1 OF 0 3

TITLE (4)
Reactor Scrams (Engineered Safety Feature Actuation)

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
										0 5 0 0 0
0 1	1 6	8 5	8 5	— 0 1 0	—	0 2	1 4	8 5		0 5 0 0 0

OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)			
POWER LEVEL (10)	0 6 1	20.402(b)	20.405(c)	X 50.73(a)(2)(iv)	73.71(b)
		20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(c)
		20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	OTHER (Specify in Abstract below and in Text, NRC Form 366A)
		20.405(a)(1)(iii)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
		20.405(a)(1)(iv)	50.73(a)(2)(ii)	50.73(a)(2)(viii)(B)	
		20.405(a)(1)(v)	50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER AREA CODE
T. L. Elton, Acting Superintendent of Regulatory Compliance	9 1 2 3 6 7 4 7 8 5 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
N/A									

SUPPLEMENTAL REPORT EXPECTED (14)

		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

On 01/16/85, at approximately 0955 CST, with the reactor mode switch in the run position and reactor power at 1496 MWt (approximately 61% power), Unit 1 received a reactor scram on a reactor vessel low water level signal.

This event was the result of a vital AC power supply trip, which caused both "A" and "B" Reactor feedpumps to runback. Consequently, a low water level condition resulted.

At approximately 1102 CST, while plant personnel were lowering reactor pressure, the reactor vessel water level increased to the high level setpoint and the reactor feedpumps tripped. Before plant personnel could reset the feedpumps, with reactor power at 0 MWt , the reactor level dropped to the low level setpoint. This caused an actuation of the Reactor Protection System logic (i.e., not a scram because all of the control rods were still inserted after the 0955 CST scram), and a Group 2 isolation. Plant personnel restored the reactor level to normal via the "A" reactor feedpump and the booster pumps as the vessel continued to depressurize.

No actual or potential safety consequences or implications resulted from these events. These events had no impact on any other Unit 1 system or on Unit 2. The health and safety of the public were not affected by these events. These are non-repetitive events; however, the last reactor scram is referenced in LER 50-321/1985-01.

A-177

6

NRC Form 366 (9-83)

NRC Form 366A (9-83)

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/85

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
EDWIN I. HATCH, UNIT I	0 5 0 0 0 3 2 1	8 5	— 0 1 0	— 0 0	0 2 OF 0 3

TEXT (If more space is required, use additional NRC Form 366A's) (17)

This 30 day LER is required by 10CFR50.73(a)(2)(iv) due to the reactor scram and actuation of other Engineered Safety Features.

On 01/16/85, at approximately 0955 CST, with the reactor mode switch in the Run position and reactor power at 1496 MWt (approximately 61% power), Unit 1 scrammed due to a reactor low water level trip signal.

While Vital AC was being powered by its alternate power supply, the alternate power supply tripped, causing a runback of both reactor feedpumps. Reactor water level then decreased to approximately -100 inches (reference instrument zero). A group 2 isolation and a reactor scram were received due to the low water level signal. HPCI and RCIC auto-started and injected to recover reactor water level; at the same time, the "A" reactor feedpump was started manually. As the feedpump was recovered, HPCI and RCIC were secured. The high water level turbine trip was received immediately thereafter for the following: the HPCI, RCIC, and reactor feedpump turbines. Reactor pressure remained steady at approximately 980 PSIG and was controlled with the EHC system (the MSIV's were not closed, and the turbine had been manually tripped). Standby gas treatment initiated on low water level; no other ECCS systems started nor were needed.

After the group 2 isolation was reset (at approximately 1005 CST), the RWCU outboard isolation valve (G31-F004) was opened to reestablish RWCU system flow in order to accurately determine the bottom head temperature (done to determine if a reactor recirculation pump could be restarted). Plant personnel then noted that the water entering the vessel had cooled down the bottom head temperature such that the differential temperature between the steam dome and the bottom head was greater than the 145° F limit of Tech. Specs. section 3.6.E. Thus, the reactor recirculation pumps could not be started. Plant personnel then decided to depressurize the vessel (by lowering the EHC setpoint) to get the temperature within the limit so that a reactor recirculation pump could be started.

At approximately 1102 CST, while plant personnel were lowering reactor pressure, the reactor vessel water level increased to the high level setpoint and the reactor feedpumps tripped. Before plant personnel could reset the feedpumps, with reactor power at 0 MWt , the reactor level dropped to the low level setpoint. This caused an actuation of the Reactor Protection System logic (i.e., not a scram because all of the control rods were still inserted after the 0955 CST scram), and a Group 2 isolation. Plant personnel restored the reactor level to normal via the "A" reactor feedpump and the booster pumps as the vessel continued to depressurize.

HNP 4,6 Letter Packages

A-178

7

NRC Form 366A (9-83)

NRC Form 366A (9-83)

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/85

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
EDWIN I. HATCH, UNIT I	0 5 0 0 0 3 2 1	8 5	— 0 1 0	— 0 0	0 3 OF 0 3

TEXT (If more space is required, use additional NRC Form 366A's) (17)

At the time of the initial scram, plant personnel noted that several other loads (i.e., in addition to Vital AC) were lost from the 600 volt bus "1C". These included the following: the "1A" RBCCW pump, the "1A" air compressor, the main turbine motor suction pump and turning gear oil pump, and an electric fire pump lock out relay on the "E" 4160 volt bus was energized. Upon a subsequent investigation, plant personnel determined that undervoltage relay ETR1A would have caused the loss of all those components, had it actuated. It is believed that the actuation of this relay was the most likely cause of the scram.

Other problems experienced during the scrams were as follows:

1. At approximately 1030 CST, RCIC was declared inoperable due to its trip and throttle valve not engaging the reset circuitry and not opening.

2. At approximately 1320 CST, HPCI was declared inoperable due to its turbine stop valve binding in the mid-position; this binding was the cause of erratic HPCI flow which was experienced during the initial scram.

3. When both HPCI and RCIC were determined to be inoperable, plant personnel depressurized the reactor with the objective of decreasing reactor pressure below 113 psig as required by Tech. Specs. sections 3.5.D.7 and 3.5.E.3.

Plant personnel investigated the RCIC problem, and determined that RCIC's trip and throttle valve was in the continuous trip position due to the centrifugal trip weight's spring becoming loose. The spring apparently became loose during operation, thus causing damage to other parts of the trip/reset mechanism. The trip and throttle valve's trip/reset mechanism was repaired by replacing the spring, spring seat, weight, tappet guide, tappet and ball assembly, and the compression spring and emergency tappet nut. RCIC was functionally tested satisfactorily per the "RCIC PUMP OPERABILITY" procedure (HNP-1-3405), and returned to service on 1/18/85.

Plant personnel investigated the HPCI problem, and determined that its turbine stop valve had stuck in the mid-position due to its having a galled stem. The valve was repaired by polishing the inside of the valve and the valve's stem. The valve was then functionally tested satisfactorily, and HPCI was returned to service on 01/17/85.

No actual or potential safety consequences or implications resulted from these events. These events had no impact on any other Unit 1 system or on Unit 2. The health and safety of the public were not affected by these events. These are non-repetitive events; however, the last reactor scram is referenced in LER 50-321/1985-03.

A-179

8

NRC Form 366A (9-83)

NRR STAFF PRESENTATION TO THE ACRS

SUBJECT: BROWNS FERRY RHR VALVE FAILURES OF JANUARY 30, 1985

DATE: MARCH 5, 1985

PRESENTER: BILL LONG

PRESENTER'S TITLE/BRANCH/DIV: PROJECT MANAGER
OPERATING REACTORS BRANCH 2
DIVISION OF LICENSING

PRESENTER'S NRC TEL. NO.: 492-7270

SUBCOMMITTEE: REACTOR OPERATIONS

A-100

9

RHR VALVE FAILURE
EVENT

FACILITY: Browns Ferry

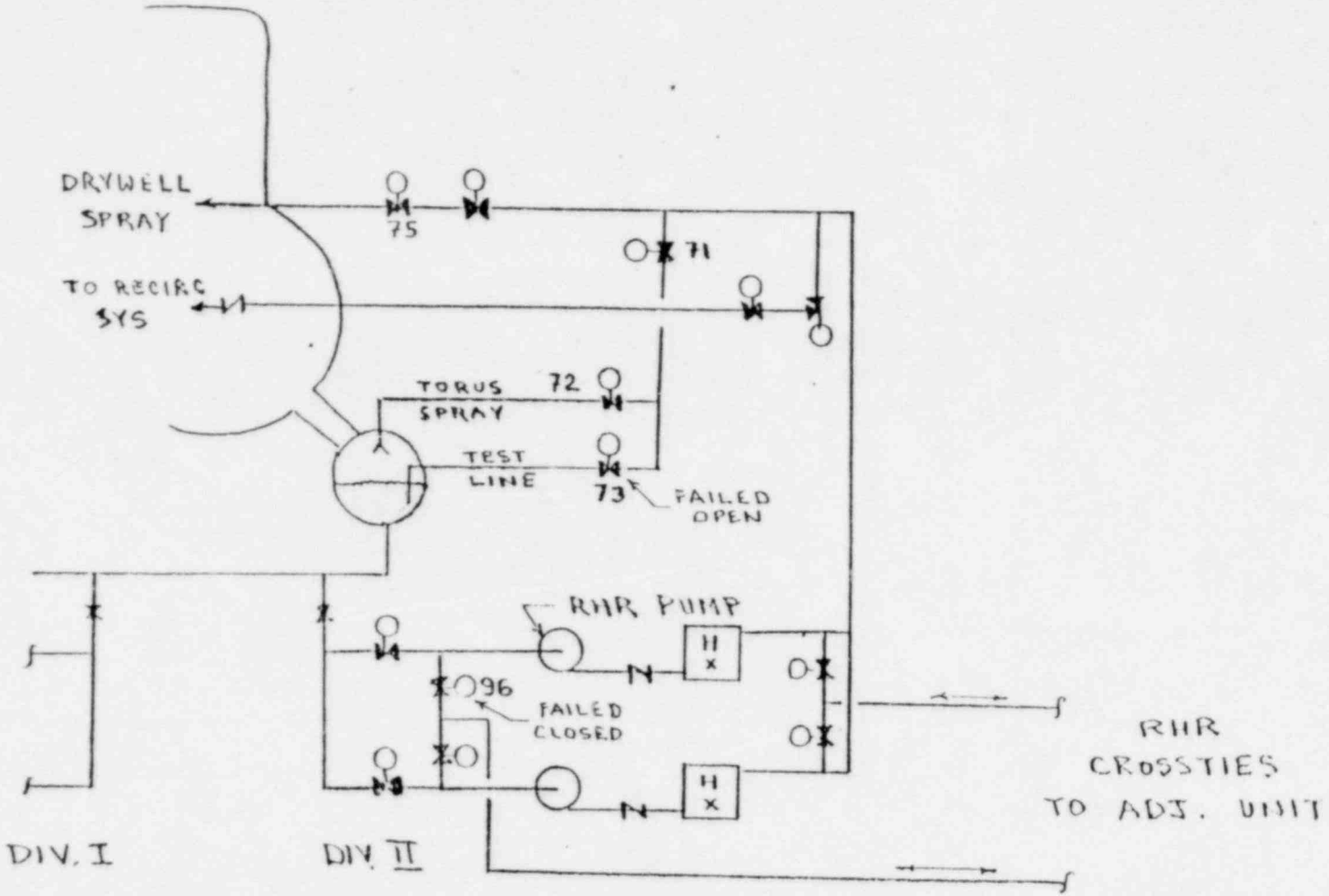
DATE: Jan 30, 1985

SOURCE: Region II Morning Report of Feb. 1, 1985

DESCRIPTION:

"Residual heat removal (RHR) valves fail to operate: on 1/30, at 4:20 P.M., after performing a vibration test on the Unit 3 RHR pump, RHR valve 73 in the RHR test line failed to shut electrically. The 73 valve also could not be shut ~~electrically~~ mechanically. The licensee discovered that the end bell of the 73 valve had come off. The licensee has repaired the valve and is evaluating the failure.

On 1/30, at 11:45 P.M., Unit 1 RHR suction cross-connect valve (96) would not operate electrically. An electrical problem existed at the local control station. Valve 96 also could not be operated mechanically due to engagement finger misadjustment. The licensee has corrected the mechanical problem and continues to troubleshoot the electrical problem. Routine followup."



RHR SYSTEM

A-182

11

JANUARY 30, 1985 RHR VALVE FAILURES
BROWNS FERRY NUCLEAR PLANT

Unit: Browns Ferry 3

Plant Status at Time of Event: Power operation

Failed Valve ID: 3-FCV-74-73

Function:
"RHR Test Valve" (RHR full-flow test discharge to torus)

Location: Torus compartment

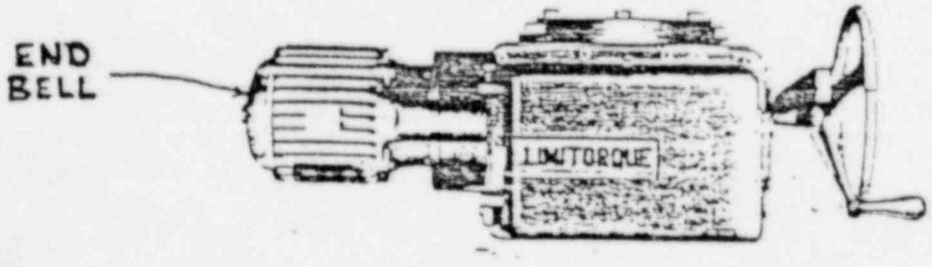
Valve Size and Type:
12-inch gate valve, installed between reducers in a 18-inch line

Operator Type: Limitorque SMB-2.

Valve Controls:
These valves, two per unit, automatically close (if open) on a LPCI signal. Interlocks then prevent reopening prior to core reflood. Valves are manually controlled from main control room and local control panel.

Failure:
Flow-induced vibration during RHR torus cooling mode operation caused loosening and breakage of end-bell bolts (four 1/4 inch bolts) on valve operator motor. End-bell fell off. Valve became electrically inoperable. Valve failed in open position. Normal position is closed.

Safety Functions Affected:
With valve 74-73 electrically inoperable, if a LOCA had occured during the torus cooling operation, the valve would have failed to close as the system automatically realigned to the LPCI mode. However, valve 73-71 would have closed thereby preventing LPCI injection flow from being diverted to the torus.
There was no loss of safety function.



Unit: Browns Ferry 2

Status at Time of Event: Refueling outage

Failed Valve ID: 2-FCV-74-96

Valve Function:
RHR Suction Crosstie (Crossconnect from Unit 2 RHR Pump A Suction Line to Unit 1 Pumps B & D Suction Line).

Location: Secondary containment

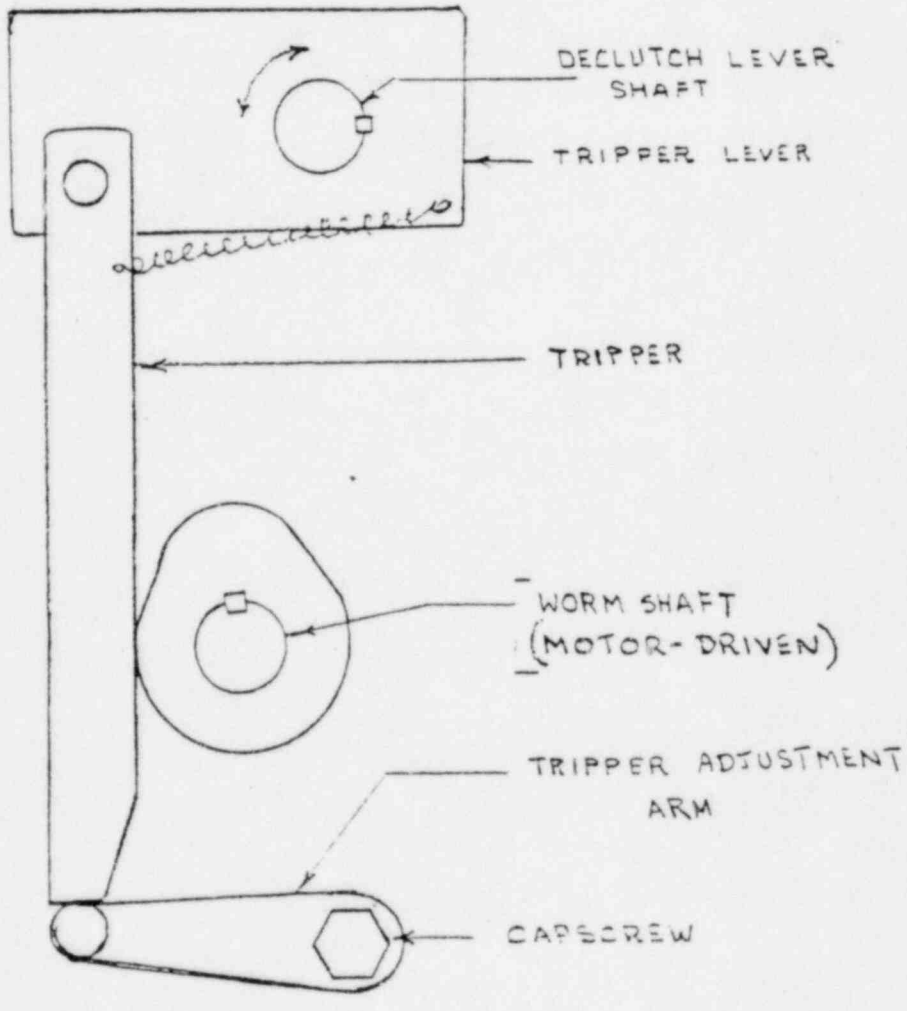
Valve Size and Type: 14-inch gate

Operator: Limitorque SMB-00

Controls:
These valves, two per plant in Units 1 & 3, and four in Unit 2, are controlled from the main control room and local control panel. The controls are normally kept denergized to prevent inadvertant opening.

Failure:
Valve became inoperable in the electrical mode due to corroded terminal blocks and switches at the local control station. Grounding at the local control station caused fuses to blow preventing electrical operation. The valve was also found to be inoperable in the handwheel mode. The tripper adjustment arm in the clutch tripper mechanism had loosened preventing the clutch from being latchable in the handwheel mode. The valve failed in the closed position.

Safety Functions Affected:
Tech Specs for each Browns Ferry plant require RHR crosstie operability to the adjacent plant. This put Unit 1 into an LCO action statement. RHR crosstie capability is not part of the safety design basis. No loss of safety function occurred. TVA has requested Tech Spec amendments to delete these LCOs and their associated surveillance requirements.



LIMITORQUE SMB-00 CLUTCH
TRIPPER MECHANISM

14

A-185

IE STAFF PRESENTATION

TO THE ACRS

SUBJECT: CATAWBA UNIT I - BOTH SAFETY INJECTION PUMPS INOPERABLE

DATE: MARCH 5, 1985

PRESENTER: ERIC WEISS for I. VILLALVA

PRESENTER'S TITLE/BRANCH/DIVISION: SENIOR REACTOR SYSTEMS ENGINEER
EVENTS ANALYSIS BRANCH, DEPER

PRESENTER'S TELEPHONE NO.: (301) 492-9007

SUBCOMMITTEE: REACTOR OPERATIONS

A-186

15

CATAWBA UNIT 1 - BOTH SAFETY INJECTION (SI) PUMPS INOPERABLE

FEBRUARY 7, 1985 (I. VILLALVA, IE)

- PLANT STATUS: OPERATING MODE 2 (STARTUP MODE, ZERO POWER).
- PROBABLE CAUSE: INADEQUATE OPERATOR TRAINING.
- SEQUENCE OF EVENTS:

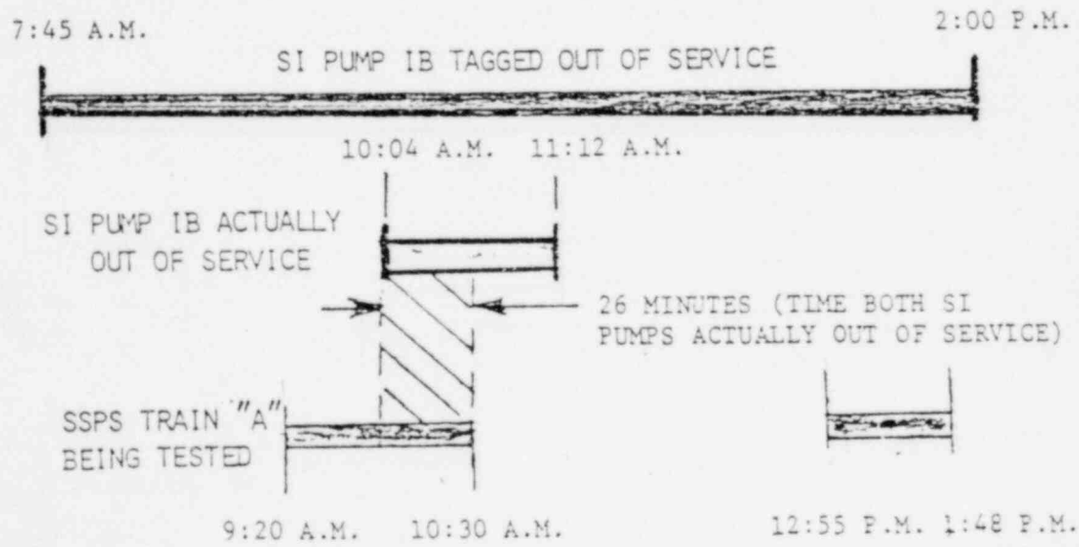
7:45 A.M. - SI PUMP IB TAGGED OUT OF SERVICE FROM 7:45 A.M. TO PAST 2:00 P.M. (NOTE: SI PUMP IB WAS ACTUALLY OUT OF SERVICE FROM 10:04 A.M. TO 11:12 A.M.)

9:20 A.M. - SHIFT SUPERVISOR AUTHORIZES BI-MONTHLY TESTING OF TRAIN "A" SSPS. (TEST LASTS TO 10:30 A.M. MAKING BOTH SI PUMPS INOPERABLE FOR 70 MINUTES.)

12:55 - SHIFT SUPERVISOR AGAIN AUTHORIZES TESTING OF TRAIN "A" SSPS. (TEST TERMINATED AT 1:48 P.M. MAKING BOTH SI PUMPS INOPERABLE FOR 53 MINUTES.)

1:48 P.M. - SHIFT SUPERVISOR INFORMED THAT BOTH SI PUMPS ARE INOPERABLE, AT WHICH TIME SSPS TESTING IS TERMINATED.

- IMMEDIATE CORRECTIVE ACTION: TERMINATING SSPS TESTS AT 1:48 P.M.
- FOLLOW-UP CORRECTIVE ACTIONS - OPERATIONS SUPERINTENDENT INIATED TRAINING ON SHIFT CHANGE TO INFORM ALL OPERATING PERSONNEL OF THE EVENT.
- SAFETY CONSEQUENCES: BOTH SI PUMPS INOPERABLE FOR 2 HRS-3MIN. PER T.S., BUT ACTUALLY UNAVAILABLE FOR 26 MINUTES, DURING WHICH TIME THE SI PUMPS WOULD NOT BE IMMEDIATELY AVAILABLE IF A LOCA WERE TO OCCUR.



IE STAFF PRESENTATION

TO THE ACRS

SUBJECT: REACTOR PROTECTION SYSTEM MALFUNCTION AT SEQUOYAH UNIT 2

DATE: MARCH 5, 1985

PRESENTER: ERIC WEISS for I. VILLALVA

PRESENTER'S TITLE/BRANCH/DIVISION: SENIOR REACTOR SYSTEMS ENGINEER
EVENTS ANALYSIS BRANCH, DEPER

PRESENTER'S TELEPHONE NO.: (301) 492-9007

SUBCOMMITTEE: REACTOR OPERATIONS

A-188

17

REACTOR PROTECTION SYSTEM MALFUNCTION AT SEQUOYAH UNIT 2 - JANUARY 12, 1985

(I. VILLALVA, IE)

- PLANT STATUS - PLANT OPERATING AT 96% POWER PRIOR TO THE EVENT.
- EVENT - RTB-A FAILS TO OPEN AUTOMATICALLY UPON A VALID REACTOR TRIP SIGNAL.
- CAUSE(S) - INADEQUATE MAINTENANCE PROCEDURES AND POST-MAINTENANCE TESTING.
- EFFECTS - MAINTENANCE ERROR RESULTS IN A SHORT-CIRCUIT FAILURE OF TRANSISTOR Q3 IN UV OUTPUT CIRCUIT. SHORT-CIRCUIT FAILURE OF Q3, IN TURN, RESULTS IN RTB-A FAILING TO OPEN AUTOMATICALLY ON A VALID REACTOR TRIP SIGNAL.
- GENERIC CONSIDERATIONS - ALL WESTINGHOUSE DESIGNED PLANTS USING A SSPS ARE SUSCEPTIBLE TO THIS TYPE OF FAILURE. FOR EXAMPLE, NORTH ANNA 2 HAS EXPERIENCED 3 SIMILAR FAILURES, EXCEPT THAT THESE FAILURES OCCURRED DURING A PLANT OUTABE AND WERE DETECTED AND REPAIRED PRIOR TO RETURNING THE PLANT TO POWER. IN ADDITION, SIMILAR FAILURES HAVE OCCURRED AT A FOREIGN REACTOR (EXACT NUMBER UNKNOWN).
- SAFETY CONSEQUENCES - NONE IN THIS CASE SINCE THE REDUNDANT UV OUTPUT CIRCUIT FUNCTIONED PROPERLY AND SCRAMMED THE REACTOR BY OPENING RTB-B. HOWEVER, THE FAILURE OF BOTH UV OUTPUT CIRCUITS DURING POWER OPERATION WOULD PREVENT THE AUTOMATIC OPENING OF BOTH RTBs ON A VALID REACTOR TRIP SIGNAL, THEREBY EXPOSING THE PLANT TO AN ATWS EVENT. (NOTE: SUCH FAILURES NOT ONLY BLOCK THE UV TRIP FUNCTION OF THE RTBs BUT ALSO THE SHUNT TRIP FUNCTION, IF INSTALLED.)
- CORRECTIVE ACTIONS TAKEN BY NRC - REGION II REVIEWED AND APPROVED THE ACTIONS TAKEN BY SEQUOYAH SUBSEQUENT TO THE EVENT AND HAS REQUESTED NRR ASSISTANCE IN REVIEWING THE ACCEPTABILITY OF THE WESTINGHOUSE DESIGN. NRR IS REVIEWING THIS MATTER, INCLUDING ITS IMPACT ON ATWS EVENTS. IE IS PREPARING AN INFORMATION NOTICE DESCRIBING THIS EVENT AND THE NORTH ANNA 2 FAILURES.

REACTOR PROTECTION SYSTEM MALFUNCTION AT SEQUOYAH UNIT 2

- CORRECTIVE ACTION TAKEN BY LICENSEE:

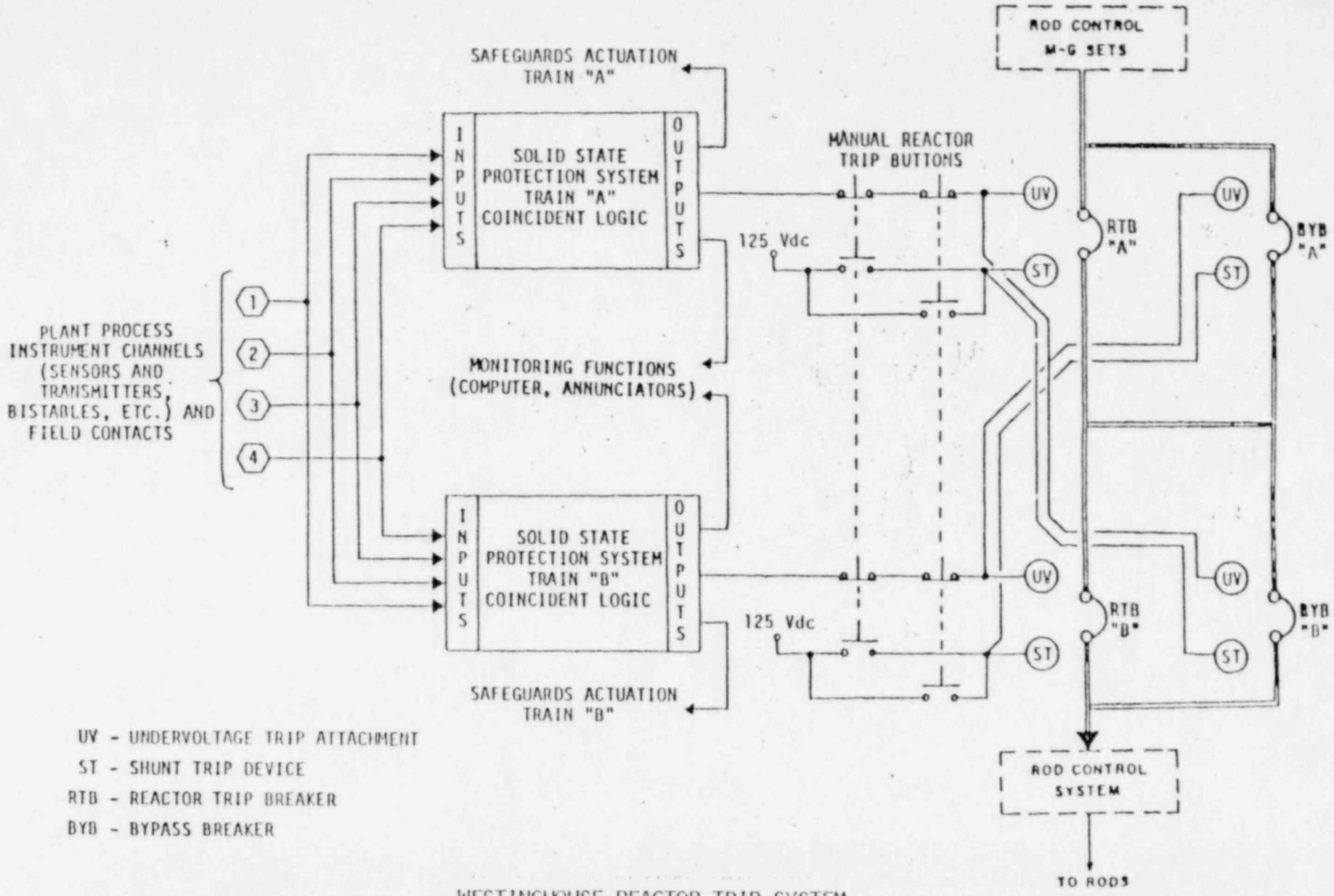
  - RTB-A MANUALLY TRIPPED: THE OPERATOR NOTICED THAT RTB-A HAD NOT OPENED ON DEMAND AND IMMEDIATELY PRESSED THE MANUAL SCRAM BUTTON WHICH OPENED RTB-A.

  - POST-TRIP REVIEW: FAILURE IS CORRECTLY DIAGNOSED AS BEING IN THE UV OUTPUT CIRCUIT OF THE SSPS. OUTPUT CIRCUIT CARD ON AFFECTED SSPS TRAIN REPLACED, AFTERWHICH THE TRIP FUNCTION OPERATED PROPERLY.

  - TESTING: EXTENSIVE TESTING OF THE RPS PERFORMED, INCLUDING RESPONSE TIME TESTS OF THE RTBs. IT WAS DETERMINED THAT THE FAILURE WAS DUE TO A SHORT-CIRCUIT IN TRANSISTOR Q3 WHICH WAS ATTRIBUTED TO IMPROPER SURVEILLANCE, I.E., THE SHORT-CIRCUIT WAS CAUSED WHILE CHECKING THE UV TRIP COIL WITH A MULTIMETER THAT WAS SET ON THE CURRENT FUNCTION RATHER THAN THE VOLTAGE FUNCTION.

  - TRAINING: MAINTENANCE AND OPERATING PERSONNEL WERE INFORMED OF THE EVENT AND ITS ROOT CAUSES.

  - PROCEDURES: MAINTENANCE, POST-MAINTENANCE AND SURVEILLANCE TEST PROCEDURES WERE REVIEWED AND REVISED TO PRECLUDE FUTURE OCCURRENCES OF SIMILAR EVENT.



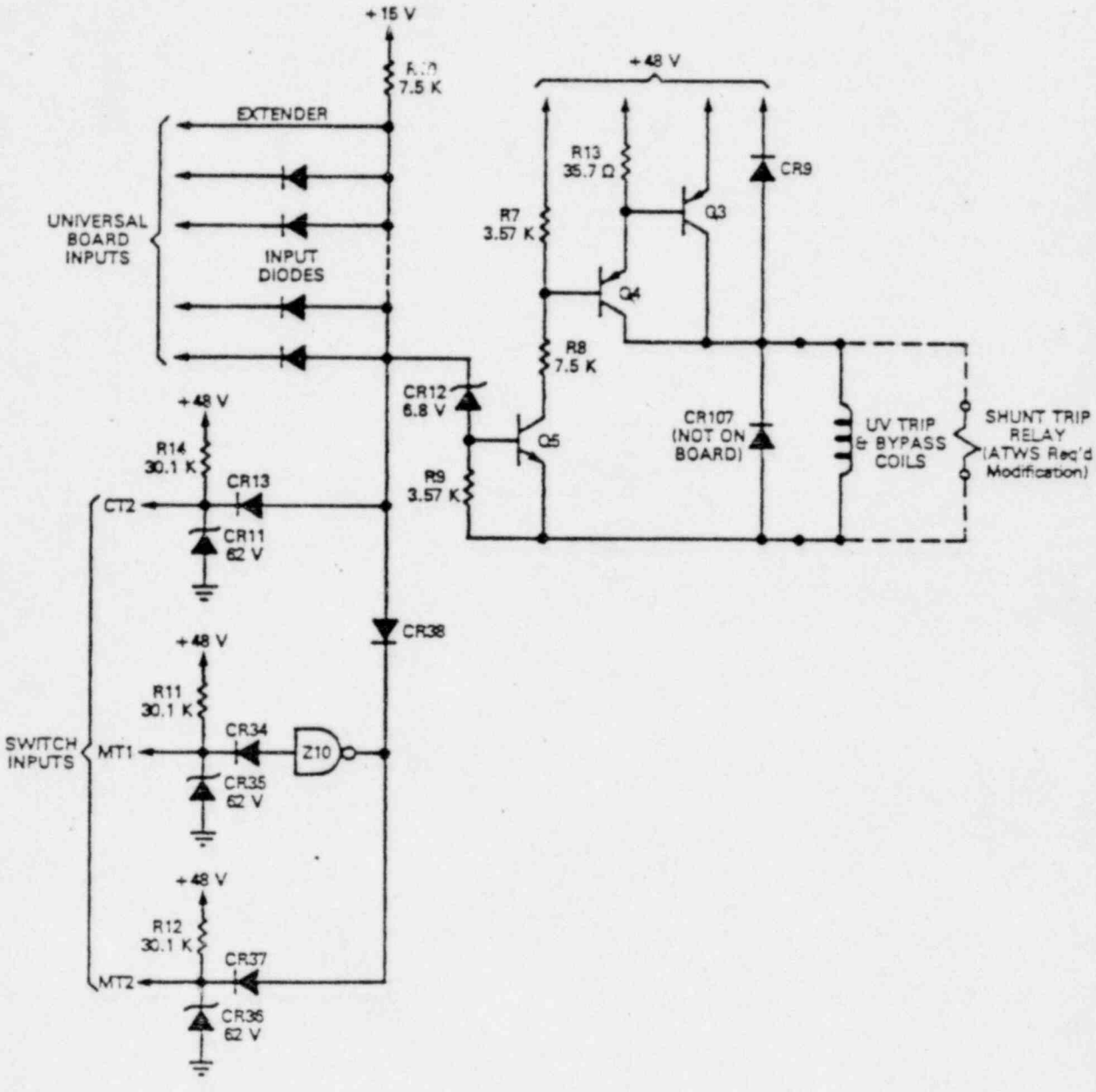
UV - UNDERVOLTAGE TRIP ATTACHMENT

ST - SHUNT TRIP DEVICE

RTB - REACTOR TRIP BREAKER

BYB - BYPASS BREAKER

WESTINGHOUSE REACTOR TRIP SYSTEM



UV OUTPUT CIRCUIT

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

March 7, 1985

IE INFORMATION NOTICE NO. 85-18: FAILURES OF UNDERVOLTAGE OUTPUT CIRCUIT BOARDS IN THE WESTINGHOUSE-DESIGNED SOLID STATE PROTECTION SYSTEM

Addressees:

All Westinghouse-designed pressurized water reactor power facilities holding an operating license (OL) or a construction permit (CP).

Purpose:

This information notice is provided to alert recipients of a potentially generic problem pertaining to short-circuit failures of the undervoltage (UV) output circuit boards in the Westinghouse-designed solid state protection system (SSPS). The failure of one UV output circuit would result in the loss of automatic reactor trip redundancy; the unavailability of both UV output circuits would result in the loss of the automatic trip function of the reactor protection system and hence make the plant susceptible to an anticipated transient without scram (ATWS) event.

It is expected that recipients will review the information contained in this notice for applicability to their facilities and consider actions, if appropriate, to preclude similar failures from occurring at their facilities. However, suggestions contained in this notice do not constitute NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances:

During a 4½-year period, three UV output circuit failures have occurred at North Anna Unit 2. As described later, Virginia Electric Power Company (VEPCO) attributes these failures to inadequate surveillance, maintenance, or modification procedures for the reactor trip breaker (RTB) switchgear cubicle associated with the failed UV output circuit. The first failure occurred in April of 1980, the second in May of 1983 and the third in October of 1984. The first failure was detected and repaired before the plant received an operating license. The second and third failures were detected and repaired during refueling outages and prior to returning to power operation. Post-maintenance testing led to the timely detection of these last two failures.

In April of 1980, a short-circuit failure was detected in the B train UV output circuit board at North Anna Unit 2. The Unit 2 maintenance history for 1980 is no longer available, thus VEPCO cannot state with certainty that this failure was maintenance related. However, because the UV coil for RTB-A was replaced

April 3, 1980 just before the B train UV output circuit failure, VEPCO indicated that it is likely that some surveillance was performed on the B train after the A train UV coil was replaced.

On May 13, 1983, a short-circuit failure was detected on the B train UV output circuit board at North Anna Unit 2. In this instance, the failure was preceded by maintenance activities on the bypass breaker for RTB-A. Because the UV trip coil for the bypass breaker for RTB-A is powered from the B train UV output circuit, VEPCO has attributed this failure to a maintenance action that inadvertently short-circuited the B train UV output circuit.

On October 29, 1984, another short-circuit failure was detected on a UV output circuit board at North Anna Unit 2. In this instance, the failure was preceded by the installation of the shunt trip relay, the relay that actuates the automatic shunt trip feature of the RTBs; therefore, VEPCO attributes this failure to inadequate plant procedures for implementing the automatic shunt trip feature of the RTBs.

On January 12, 1985, while the plant was operating at approximately 96 percent power, RTB-A failed to open automatically on a valid reactor trip signal at Sequoyah Unit 2; however, RTB-B opened automatically, causing the control rods to insert. The failure was recognized immediately by the operator who then followed up the automatic scram demand with a manual scram which opened RTB-B. This failure to trip was the direct result of a pre-existing short-circuit failure on the A train UV output circuit. Sequoyah's staff initially believed that maintenance had not been performed on external components associated with the A train UV output circuit prior to this event. The failure, therefore, was originally considered to have been a random failure rather than a maintenance-related failure. However, a subsequent investigation revealed that surveillance activities had been performed on the UV trip attachment of RTB-A prior to the event. As a result, Sequoyah management now believes that a multimeter was being used to check the voltage across the UV coil of RTB-A, and that the meter was set on a current setting rather than on a voltage setting, thereby shorting out the A train UV output circuit. In this regard, the Sequoyah failure appears to be similar to those at North Anna.

Discussion:

The UV output circuit (see Attachment 1, UV Output Circuit) is designed so that in the absence of a reactor trip signal from the universal logic boards or switched inputs, the Darlington pair transistors Q3 and Q4 in the UV output circuit will conduct current and energize the UV trip coil of the associated RTB. A reactor trip input will result in the turning off of the Darlington pair transistors Q3 and Q4, thereby interrupting current flow to the UV trip coil of the associated RTB and causing the RTB to open. Similarly, if the automatic shunt trip feature has been implemented, the turning off of transistors Q3 and Q4 would also deenergize the associated shunt trip relay, thereby tripping the same RTB by a diverse mechanism.

Failures of the type experienced at North Anna and Sequoyah have resulted in the shorting of transistors Q3 or Q4. Such short-circuit failures are not only

not fail-safe (i.e., they do not cause a reactor trip) but they also would prevent the functioning of any subsequent automatic reactor trip signal because the failure would continuously maintain power to the UV trip coil of the associated RTB. Since the Darlington pair transistors Q3 and Q4 are also being used to control the automatic shunt trip feature of the RTBs, failures of the type experienced at North Anna and Sequoyah would also incapacitate the automatic shunt trip feature of the associated RTB. Thus, the coincident failures of both the A and B train UV output circuits would disable both automatic trip functions (UV trip and shunt trip) of both RTBs, thereby exposing the plant to a potential ATWS event.

Because the UV output circuits are not continuously monitored for failures and because each UV output circuit is functionally tested on a 60-day cycle, one of the two redundant UV output circuits could be inoperable for as long as 60 days before the failure was detected. Thus, although the probability of an ATWS event occurring due to failed UV output circuit boards is very low, the most likely scenario for such an event appears to be one where a reactor trip demand occurs while one SSPS train is being tested and the other train has a short-circuited UV output circuit board. For example, if the Sequoyah Unit 2 reactor trip of January 12, 1985 had occurred while the B train SSPS was being tested, an automatic reactor trip would not have occurred.

The described failures appear to be associated with maintenance activities external to the SSPS, including the implementing of the automatic shunt trip features of the RTBs. To preclude potential failures of the UV output circuit boards while implementing the shunt trip feature, Westinghouse informed North Anna that it is imperative that the SSPS UV output circuit card be removed before performing such work. Prudent actions also would include testing the UV output circuit by using the SSPS semiautomatic testor after any maintenance work is performed on the associated RTB cubicle.

No specific action or written response is required by this information notice; however, if you have any questions regarding this notice, please contact the Regional Administrator of the appropriate NRC regional office or the technical contacts listed below.

Edward L. Jordan, Director
Division of Emergency Preparedness
and Engineering Response
Office of Inspection and Enforcement

Technical Contacts: I. Villalva, IE
(301) 492-9007

N. Merriweather, RII
(404) 221-5577

Attachments:
1. UV Output Circuit
2. List of Recently Issued IE Information Notices

NRR STAFF PRESENTATION TO THE ACRS

SUBJECT: FARLEY 2 - FAILURE OF CONTAINMENT TENDON FIELD ANCHORS

DATE: MARCH 5, 1985

PRESENTER: George Lear for EDWARD A. REEVES

PRESENTER'S TITLE/BRANCH/DIV: PROJECT MANAGER/ORB#1/DIVISION OF LICENSING

PRESENTER'S NRC TEL. NO.: 492-7386

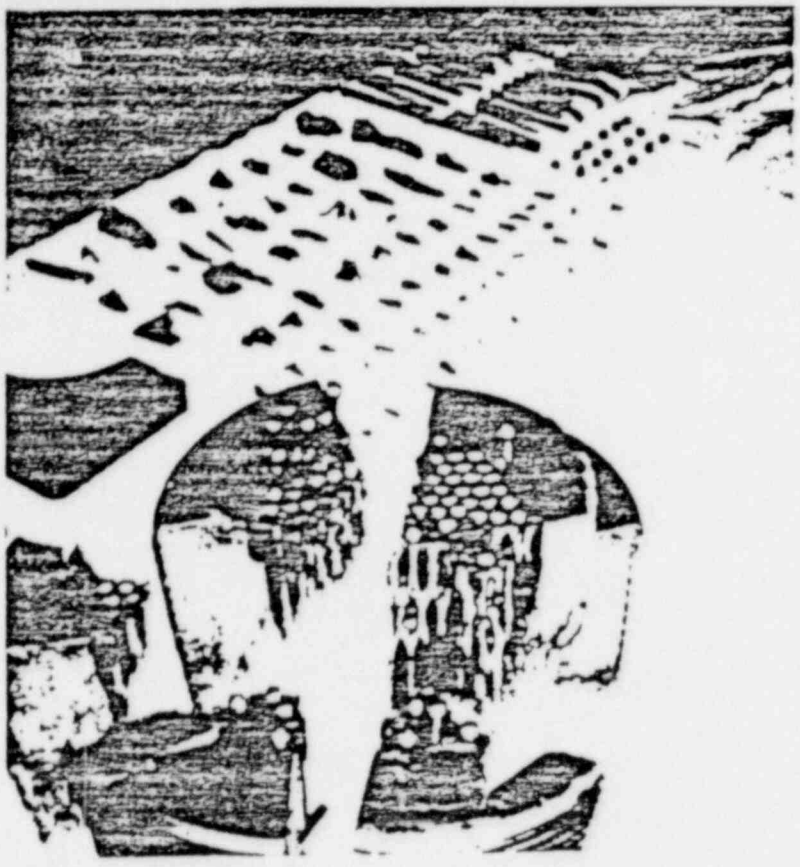
SUBCOMMITTEE: REACTOR OPERATIONS

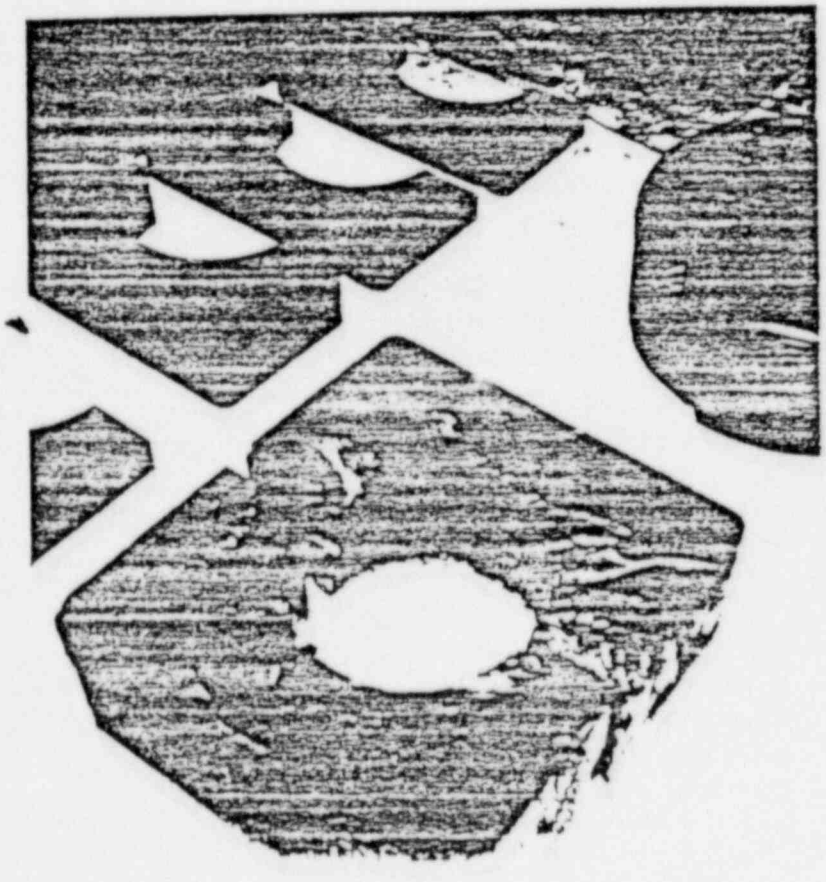
A-196

OPERATING REACTOR EVENTS BRIEFING

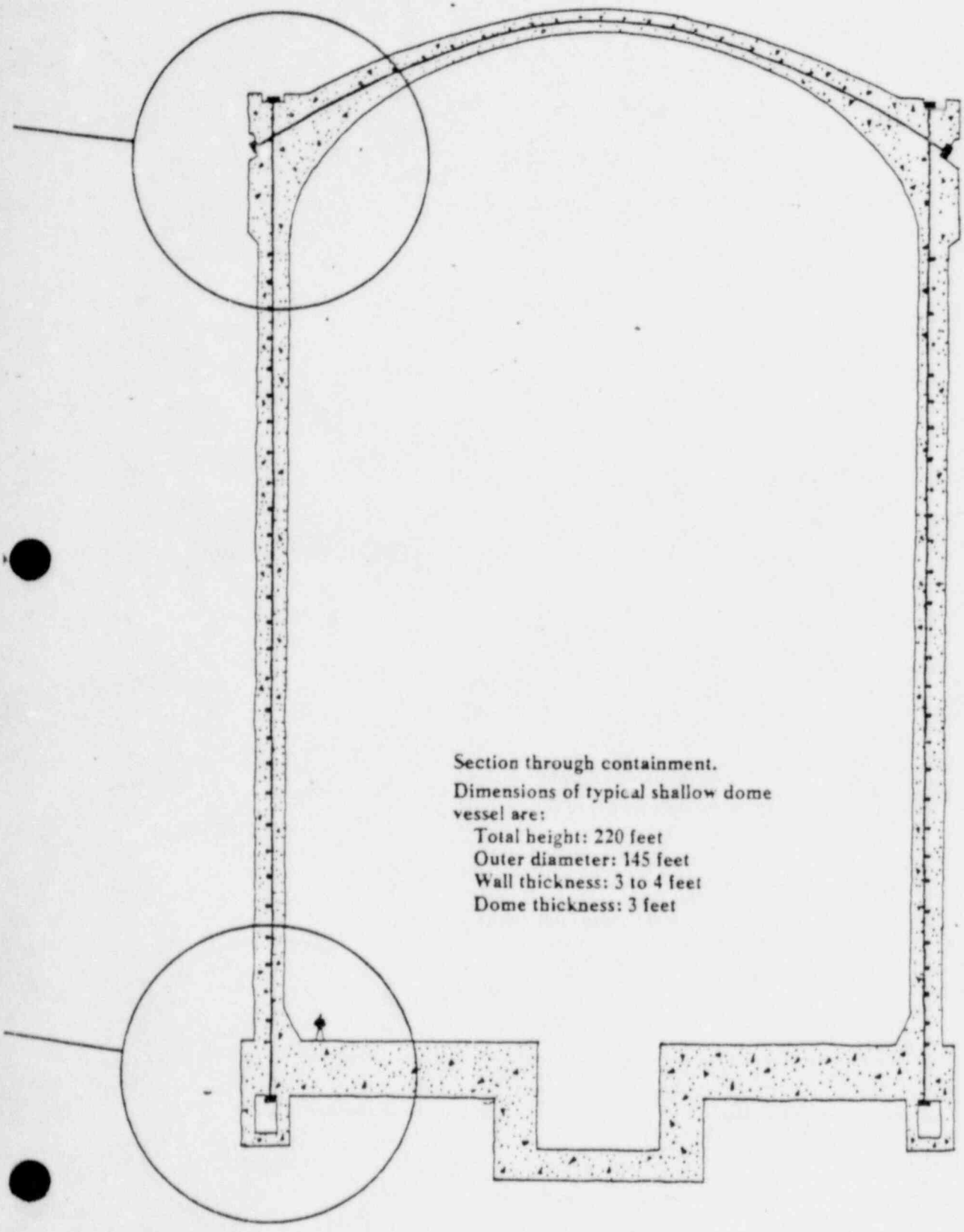
FARLEY 2 - FAILURE OF CONTAINMENT TENDON FIELD ANCHOR REPORTED ON JANUARY 28, 1985 (ED REEVES, PROJECT MANAGER, ORB#1, DL)

- PRE-ILRT CHECKS REVEALED A DAMAGED TENDON GREASE CAP
- LICENSEE INVESTIGATION REVEALED THE FOLLOWING:
- ONE TENDON DETENSIONED DUE TO FIELD ANCHOR FRACTURE AND A SECOND FIELD ANCHOR WAS CRACKED.
  - HV HEAT DESIGNATION WAS SAME ON BOTH ANCHORS.
  - CHECK OF UNIT 1 TENDON GREASE CAPS WAS SATISFACTORY.
  - APCo MET WITH NRC STAFF ON 2/7/85 TO PRESENT BACKGROUND AND PLAN OF ATTACK AS FOLLOWS:
    - ASSEMBLE WORK TEAMS AND EQUIPMENT (APCo, BECHTEL AND INRYCO).
    - 49 HV HEAT ANCHORS TO BE CHECKED AND REPLACED ON UNIT 2 (SUSPECTED TEMPERATURE EMBRITTLEMENT).
    - 55 NON - HV ANCHORS TO BE CHECKED ON UNIT 2.
    - LABORATORY ANALYSES OF GREASE AND ANCHORS STARTED.
    - 2/12/85 BEGAN INTEGRATED WORK, EVALUATION TO WORK CONTINUOUSLY
- IE NOTICE 85-10 ISSUED 2/6/85 TO ALL LICENSEES. REVIEW RESPONSIBILITY TRANSFERRED FROM REGION 2 TO NRR.
- APCo MET WITH NRC STAFF ON 3/1/85 OT DISCUSS INCREASE IN WORK SCOPE AS FOLLOWS:
  - THIRD VERTICAL FIELD ANCHOR FOUND PARTIALLY FAILED.
  - LAB RESULTS OF FAILED FIELD ANCHORS SHOW HYDROGEN STRESS CRACKING IN CENTRAL HONEYCOMB AREA.
  - MAJOR SOURCE OF HYDROGEN WAS FROM WATER REACTION WITH ZINC.
- NRC STAFF AGREED ILRT COULD SAFELY PROCEED (STARTED 3/1/85, COMPLETED 3/3/85).





A-191



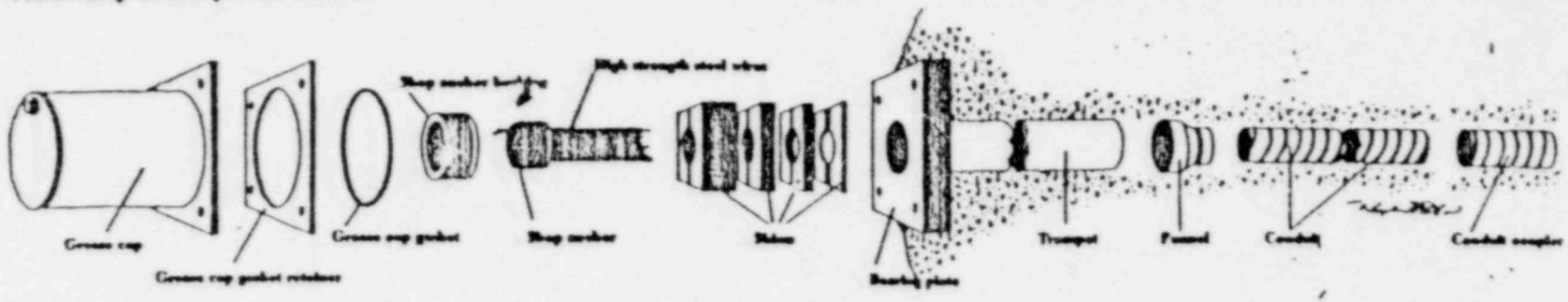
Section through containment.

Dimensions of typical shallow dome vessel are:

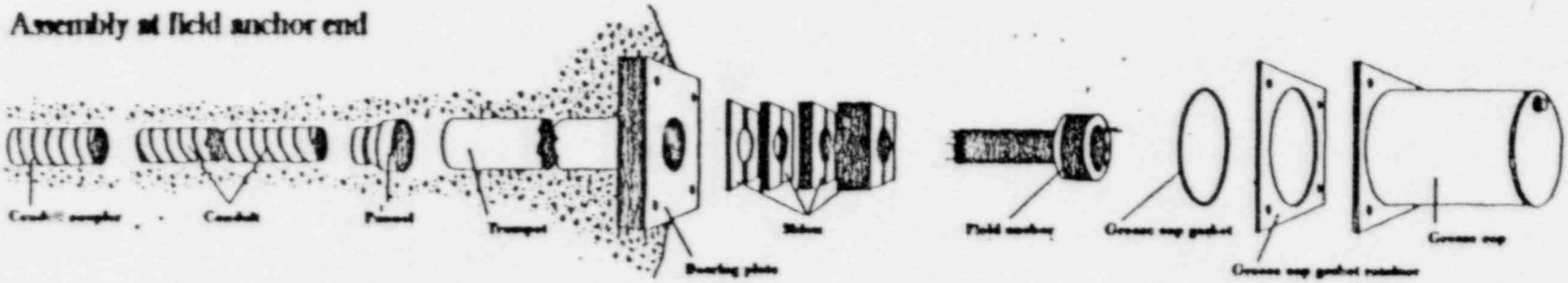
Total height: 220 feet
Outer diameter: 145 feet
Wall thickness: 3 to 4 feet
Dome thickness: 3 feet

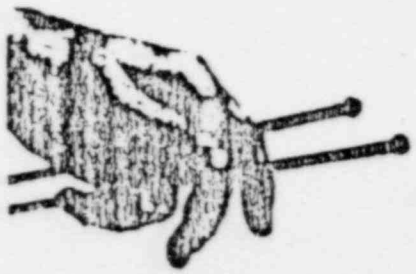
A-200

Assembly at shop anchor end



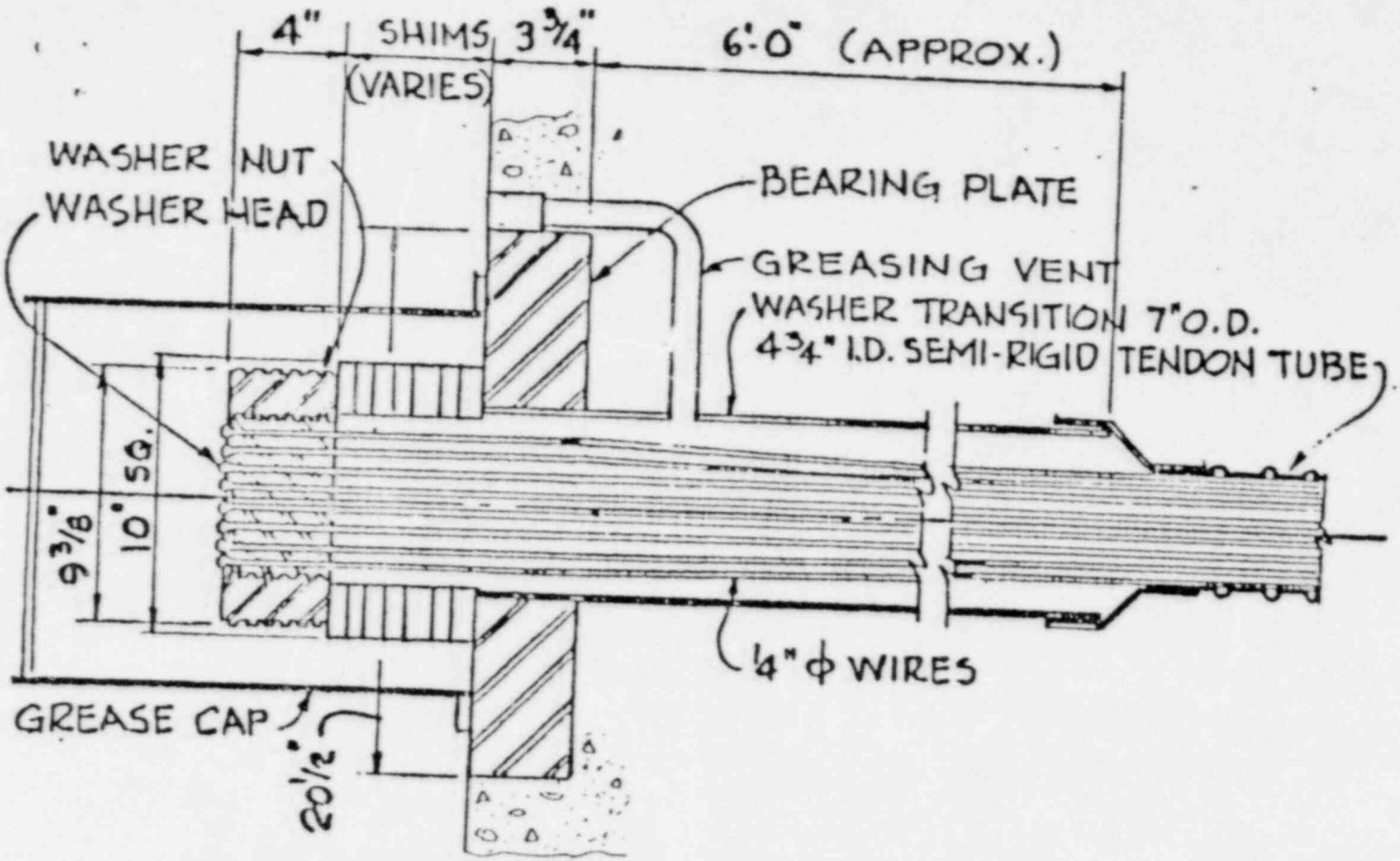
Assembly at field anchor end





A-281

30



FARLEY 170 WIRE PRESTRESS SYSTEM

NOTE: SHOP END SHOWN.
ON FIELD END WASHER HEAD AND NUT ARE AN INTEGRAL PART.

FARLEY 2 ANCHOR HEAD

INSPECTION STATUS

130 VERTICAL TENDONS:

INSPECTED	REPLACED	CONDITION
130	21	1 FAILED 1 CRACKED (FULL LOAD) 1 CRACKED (75% DETENSIONED 47 WATER-TRACES TO 1.5 GAL 66 NO WATER NOTED 10 MT SATISFACTORY 8 MT CRACKS

93 DOME TENDONS:

INSPECTED	REPLACED	CONDITION
25	0	1 VISUAL MOISTURE 24 NO WATER

134 HOOP TENDONS:

INSPECTED	REPLACED	CONDITION
63	7	2 WATER FIELD END 1 WATER SHOP END 7 MT SATISFACTORY

CONCLUSIONS

- INSPECTION PROGRAM PROVIDES ASSURANCE THAT ALL VERTICAL TENDONS HAVE NO VISIBLE CRACKS AND AT LEAST A 95% CONFIDENCE LEVEL THAT 95% OF ALL HOOP AND DOME TENDONS HAVE NO VISIBLE CRACKS
- A TOTAL OF 20 TENDONS CAN BE SACRIFICED WITHOUT AFFECTING MINIMUM INTENTED DESIGN MARGINS AT END OF 40 YEARS
- MARGINS OF AT LEAST 32% AND AS HIGH AS 45% EXIST (EXCLUDING 35 TENDONS) WITHOUT SACRIFICING STRUCTURAL INTREGRITY
- PRESSURIZATION OF CONTAINMENT INCREASES THE TENDON LOAD LESS THAN 4%
- ILRT CAN PROCEED WITHOUT AFFECTING STRUCTURAL INTEGRITY

NRR STAFF PRESENTATION TO ACRS

SUBJECT: GESSAR-II PRA

DATE: MARCH 8 1985

PRESENTER: MARK RUBIN

PRESENTER'S TITLE/BRANCH/DIV: RISK ANALYST/RRAB/DST

PRESENTER'S NRC TEL. NO.: 492-8315

ACTIONS REQUIRED BY SEVERE ACCIDENT POLICY STATEMENT INCLUDE
(NUREG 1070)

COMPLETION OF A PROBABILISTIC RISK ASSESSMENT (PRA) AND CONSIDERATION OF THE SEVERE-ACCIDENT VULNERABILITIES IT EXPOSES ALONG WITH THE INSIGHTS THAT IT MAY ADD TO THE ASSURANCE OF NO UNDUE RISK TO PUBLIC HEALTH, SAFETY, AND PROPERTY.

STAFF USE OF GESSAR-II PRA

TOOL TO AID IN SYSTEMATIC EVALUATION OF GESSAR-II DESIGN FEATURES

PROVIDE INSIGHTS INTO STRENGTHS AND VULNERABILITIES OF GESSAR-II DESIGN

ASSIST IN THE EVALUATION OF IMPACT OF SELECTED USIs AND GIs

ASSIST IN EVALUATION OF COMPARATIVE ADVANTAGES OF VARIOUS PROPOSED DESIGN MODIFICATIONS

GAIN INSIGHT OF IMPACT OF IMPROVED SOURCE TERM ANALYSIS

PRA NOT UTILIZED TO DEMONSTRATE COMPLIANCE WITH BOTTOM LINE CORE MELT OR RISK VALUES

GENERAL INSIGHTS (OVERVIEW)

REVISED CORE MELT FREQUENCY FOR INTERNAL EVENTS 4x10-5/YR (FOR NATIONAL AVERAGE SITE)

APPROXIMATELY 85 PERCENT OF CORE MELT ATTRIBUTED TO LOSS OF OFFSITE POWER TRANSIENTS

VERY LOW CONSEQUENCE ESTIMATES (EARLY & LATE FATALITIES) THROUGH APPLICATION OF PRELIMINARY ASTPO DATA

REVISED SEISMIC CORE MELT FREQUENCY 7x10-5/YR
DOMINATED BY POSTULATED SEISMIC RELAY CHATTER IMPACT

DUE TO LOW CORE MELT AND CONSEQUENCE ESTIMATES, RELATIVELY LOW INCENTIVE FOR EXPENSIVE PREVENTION AND MITIGATION DESIGN MODIFICATIONS

STAFF WORK ONGOING

DESIGN MODIFICATIONS

EVALUATION OF CONTAINMENT BYPASS SEQUENCES (SEISMIC AND RCIC BREAK)

SEISMIC CONSEQUENCE ANALYSIS

RELAY CHATTER

OBJECTIVE/SCOPE OF REVIEW

OBJECTIVE

EVALUATE GESSAR-II PRA

- COMPLETENESS OF DOMINANT ACCIDENT SEQUENCES
- QUANTIFICATION OF DOMINANT ACCIDENT SEQUENCES

SCOPE

- REVIEWED ASSUMPTIONS AND METHODS
- RE-EVALUATED PRA RESULTS WITHIN SCOPE AND FRAMEWORK OF GESSAR-II STUDY
- CONSIDERED: INITIATORS (INTERNAL & EXTERNAL), DATA, ACCIDENT SEQUENCES, CORE DAMAGE FREQUENCY AND ASSOCIATED UNCERTAINTIES
- CONTAINMENT FAILURE ANALYSES

GENERAL AREAS OF MODIFICATION TO GESSAR-II PRA

- ADDITIONAL DEPENDENCIES BETWEEN SAFETY FUNCTIONS
- MODELING OF COMMON MODE FAILURE OF THREE DGs
- FREQUENCY OF FAILURE OF REACTOR PROTECTION SYSTEM
- FREQUENCY OF TRANSIENTS
- CONTAINMENT EVENT TREES (CLASS II TRANSIENTS)
- SITE HAZARD FUNCTION
- COMPONENT FRAGILITIES
- SEISMIC SYSTEM ANALYSIS

REVISED GESSAR DOMINANT ACCIDENT SEQUENCES
INTERNAL EVENTS

SEQUENCE		CORE MELT (NATIONAL AVERAGE SITE)	
LOSS OF OFFSITE POWER	T_EUV, T_EW	3.2x10-5/YR	- 85%
ISOLATION TRANSIENTS	T_FQW, T_FQUX	2.4x10-6	- 6%
ATWS		2.5x10-6	- 7%
LOSS OF DC POWER	$T_{DC}UV$	4.3x10-7	- 1%
	TOTAL*	3.8x10-5	

* TOTAL OF ALL SEQUENCES CONSIDERED NOT JUST THE DOMINANT SEQUENCES. DOES NOT INCLUDE IMPACT OF UPPS.

TOTAL SEISMIC CORE MELT
(INCORPORATING BNL SYSTEM ANALYSIS CHANGES)

	GE FRAGILITIES	ALTERNATE FRAGILITIES (CASES)
GESSAR II HAZARD CURE	6.3x10-6	6.7x10-5
ZION	6.9x10-7	9.4x10-6
WATTS BAR (LLNL)	1.7x10-4	7.4x10-4

CONCLUSIONS

GESSAR INTERNAL EVENTS ANALYSIS RELATIVELY GOOD - RE-EVALUATION INDICATES REASONABLY LOW CM VALUES.

UPPS RESULTS IN CM ESTIMATES OF APPROXIMATELY 1x10-5/YR. (REDUCTION OF ABOUT 5)

SEISMIC EVALUATION RESULTED IN UPWARD REASSESSMENT DUE TO MODIFICATIONS

- A SEISMIC SYSTEMS ANALYSIS
- B INCORPORATION OF RELAY CHATTER
- C ALTERNATE FRAGILITY VALUES
- D SITE HAZARD

SEISMIC CM CONTRIBUTION RELATIVELY EQUIVALENT TO INTERNAL EVENTS.

SITE HAZARD AND FRAGILITY VALUES MUST BE JUSTIFIED ON PLANT APPLICATION BASIS.

DESIGN MODIFICATIONS - BYPASS SEQUENCES AND SEISMIC CONSEQUENCES STILL BEING EVALUATED.

GENERAL ELECTRIC COMPANY

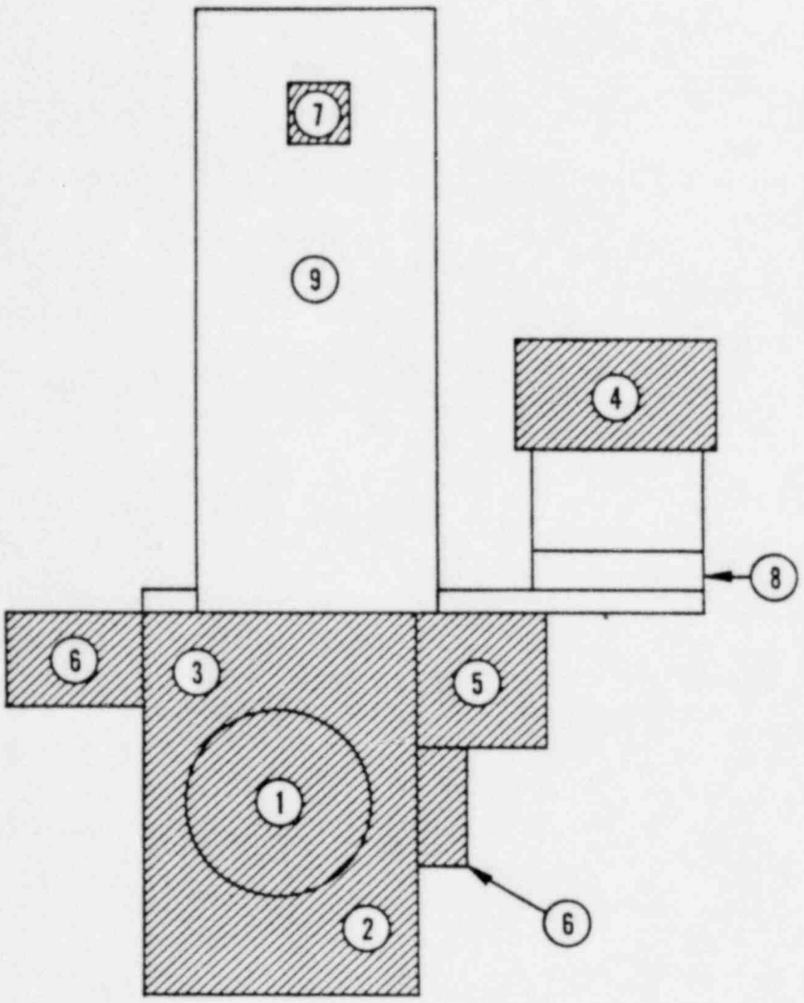
GESSAR DESCRIPTION

AND

COMPARISON WITH EXISTING PLANTS

R. VILLA
SAFETY & LICENSING OPERATIONS
MARCH 8, 1985

STANDARD BWR/6 MARK III
GESSAR II NUCLEAR ISLAND ARRANGEMENT



NUCLEAR ISLAND

1. REACTOR BUILDING
2. FUEL BUILDING
3. AUXILIARY BUILDING
4. RADWASTE BUILDING
5. CONTROL BUILDING
6. DIESEL GENERATOR BUILDING
7. OFFGAS SYSTEM

BALANCE OF PLANT

8. SERVICE FACILITIES
9. TURBINE ISLAND

- o BWR/6 MARK III
- o SITING ENVELOPE INCLUDED
  - METHODOLOGY
  - HYDROLOGY
  - SEISMOLOGY AND GEOLOGY
- o APPLICABLE TO 90% OF POTENTIAL U.S. SITES

COMPARISON BETWEEN PRINCIPAL DESIGN FEATURES OF GESSAR AND OTHER BWR FACILITIES

DESIGN FEATURES	GESSAR	PERRY	CLINTON	GRAND GULF
RATED POWER (MWT)	3579	3579	2894	3833
MAIN STEAM LINE DIAMETER (IN)	26	26	24	28
STEAM BYPASS CAPACITY (% DESIGN STEAM FLOW)	35	35	35	35
REACTOR COOLANT RECIRCULATION PIPING DIAMETER (IN)	22/24	24	20	24
REACTOR VESSEL INSIDE DIAMETER (IN)	238	238	218	251
REACTOR VESSEL INSIDE HEIGHT (FT)	70.33	70.33	69.3	73
NUMBER OF FUEL ASSEMBLIES	748	748	624	800

9/12-4

COMPARISON BETWEEN PRINCIPAL DESIGN FEATURES
OF GESSAR AND OTHER BWR FACILITIES

DESIGN FEATURES	GESSAR	PERRY	CLINTON	GRAND GULF
CONTAINMENT TYPE	o MARK III o FREE STANDING o REINFORCED CONCRETE SHIELD BUILDING	o MARK III o FREE STANDING o REINFORCED CONCRETE SHIELD BUILDING	o MARK III o STEEL LINED o REINFORCED CONCRETE	o MARK III o STEEL LINED o REINFORCED CONCRETE
CONTAINMENT DESIGN PRESSURE (PSIG)	15*	15	15	15
CONTAINMENT LEAKAGE RATE (%/DAY)	1.0	0.20	0.65	0.35
SUPPRESSION POOL	STEEL LINED REINFORCED CONCRETE	STEEL LINED REINFORCED CONCRETE	STEEL LINED CONCRETE	STEEL LINED REINFORCED CONCRETE
NUCLEAR SYSTEM PROTECTION SYSTEM	SOLID-STATE	RELAY	SOLID-STATE	RELAY

* 45 PSIG LEVEL C

COMPARISON BETWEEN SITING PARAMETERS OF GESSAR AND OTHER BWR FACILITIES

PARAMETERS	GESSAR	PERRY	CLINTON	GRAND GULF
OBE				
HORIZONTAL (G)	0.15	0.075	0.10	0.075
VERTICAL (G)	0.10	0.075	0.10	0.05
SSE				
HORIZONTAL (G)	0.30	0.15	0.25	0.15
VERTICAL (G)	0.20	0.15	0.25	0.10
MAXIMUM SUSTAINED WIND (MPH)	130	90	85	90
TORNADO				
TRANSLATIONAL (MPH)	70	70	70	60
TANGENTIAL (MPH)	290	290	290	300

A.2-14

ADDITIONAL GESSAR FEATURES

- DESIGN CHANGES REQUIRED BY CP RULE
  - HYDROGEN CONTROL SYSTEM
  - INCREASED CONTAINMENT STRENGTH
  - PROVISIONS FOR CONTAINMENT VENTING

- DESIGN CHANGES MADE AFTER PRA
  - ULTIMATE PLANT PROTECTION SYSTEM
    - CORE MAKEUP WATER
    - VESSEL DEPRESSURIZATION
    - CONTAINMENT VENTING
  - PRESENTED AS AN ALTERNATE TO HYDROGEN CONTROL

APPENDIX XXII
GE COMPANY PRESENTATION ON THE GESSAR PRA

GENERAL ELECTRIC COMPANY

GESSAR PROBABILISTIC RISK ASSESSMENT

R. VILLA
SAFETY & LICENSING OPERATION

MARCH 8, 1985

GESSAR LICENSING ACTIVITY

O APPLICATION FOR PRELIMINARY DESIGN APPROVAL	APR-1973
O ACRS REVIEW	JUL-1974 MAR-1975
O GESSAR PDA #1 ISSUED	DEC-1975
O APPLICATION FOR FINAL DESIGN APPROVAL	MAR-1980
O ACRS REVIEW	APR-1983 JUN-1983
O GESSAR FDA #1 ISSUED	JUL-1983
O NRC SEVERE ACCIDENT POLICY DRAFTED	JAN-1982
O SEVERE ACCIDENT PRA SUBMITTED	MAR-1982
O SEVERE ACCIDENT SER ISSUED	NOV-1984
O ACRS SEVERE ACCIDENT REVIEW	OCT-1984 PRESENT

NRC POLICY ON FUTURE REACTOR DESIGNS

- COMPLIANCE WITH CURRENT COMMISSION REGULATIONS
  - INCLUDING TMI REQUIREMENTS DEFINED IN THE CP RULE
  - INCLUDING A HYDROGEN CONTROL SYSTEM
  - INCLUDING A STRONGER CONTAINMENT
  - INCLUDING PROVISIONS FOR CONTAINMENT VENTING

- RESOLUTION OF UNRESOLVED AND GENERIC SAFETY ISSUES
  - INCLUDING A FOCUS ON THE RELIABILITY OF DECAY HEAT REMOVAL SYSTEMS
  - INCLUDING A FOCUS ON THE RELIABILITY OF AC AND DC ELECTRICAL SUPPLY SYSTEMS

- COMPLETION OF A PROBABILISTIC RISK ASSESSMENT
  - CONSIDERATION OF SEVERE ACCIDENT VULNERABILITIES AND THEIR ADDED IMPACT ON THE PUBLIC

- COMPLETION OF NRC STAFF REVIEW OF THE DESIGN
  - WITH A CONCLUSION OF SAFETY ACCEPTABILITY STRESSING DETERMINISTIC ENGINEERING ANALYSIS USING JUDGEMENT COMPLEMENTED BY A PRA

PURPOSE OF THE PRA

- A PRA IS INTENDED TO PREDICT RISKS THROUGH A SYSTEMATIC REVIEW OF DESIGN AND OPERATION OF A POWER PLANT

  IT PROVIDES AN INTEGRATED ASSESSMENT OF CORE MELT, CONTAINMENT FAILURE, AND RADIOLOGICAL CONSEQUENCES

  IT IS NOT CONSTRAINED TO HIGHLY PRESCRIBED DESIGN BASIS EVENTS (THE DETERMINISTIC APPROACH)

- A PRA HAS SOME LIMITATIONS

  HOW TO QUANTIFY SABOTAGE ATTEMPTS

  HOW TO QUANTIFY UNDETECTED DESIGN ERRORS

  HOW TO QUANTIFY THE LIKELIHOOD OF OPERATOR ERRORS

- SUCH LIMITATIONS MAKE PRA's UNRELIABLE AT PREDICTING THE PRECISE MAGNITUDE OF RISK

  A PRA CAN PREDICT MANY WAYS A REACTOR MIGHT BE VULNERABLE TO SEVERE ACCIDENTS

  A PRA CAN ESTIMATE THE IMPORTANCE OF SAFETY ISSUES

  A PRA CAN PUT REACTOR SAFETY ISSUES IN CONTEXT

  A PRA CAN PROVIDE A NUMERICAL ESTIMATE OF EXISTING UNCERTAINTIES IN REACTOR SAFETY

PURPOSE OF THE PRA

- OBJECTIVELY A PRA MAY IDENTIFY PLANT FEATURES WHICH MERIT CLOSE ATTENTION OR POINT OUT UNNECESSARY REGULATIONS

- HOWEVER A PRA SHOULD BE USED AS AN EXTENSION RATHER THAN A REPLACEMENT OF A DETERMINISTIC EVALUATION

THE FOCUS SHOULD BE ON PLANT VULNERABILITIES NOT CONSIDERED DURING THE DETERMINISTIC REVIEW

GESSAR PRA SCOPE

- O INTERNAL EVENTS ANALYSIS
  - LEVEL III PRA AS DEFINED IN PRA PROCEDURES GUIDE
  - COMPREHENSIVE, STATE-OF-THE-ART ANALYSIS
  - INCLUDED UNCERTAINTY ANALYSIS

- O EXTERNAL EVENTS ANALYSIS
  - SEISMIC
  - INTERNAL FIRES AND FLOODS
  - QUALITATIVE EVALUATION OF OTHER EXTERNAL EVENTS

- O DEFINITION OF PLANT & SITE
  - BWR/6 MARK III NUCLEAR ISLAND
  - ATLANTIC COASTAL SITE

- O ADDITIONAL SEVERE ACCIDENT EVALUATIONS
  - EVALUATED SABOTAGE
  - EVALUATED PROPOSED DESIGN MODIFICATIONS
  - RESOLVED APPLICABLE UNRESOLVED AND GENERIC SAFETY ISSUES

GESSAR

PRINCIPAL PRA RESULTS

- EVALUATION AGAINST INTERIM SAFETY GOAL

  CORE DAMAGE FREQUENCY: 1×10^{-4}/YR

INTERNAL EVENTS

CORE DAMAGE FREQUENCY: 4.7×10^{-6}/YR

EXTERNAL EVENTS

SEISMIC ANALYSIS CORE DAMAGE FREQUENCY: 4.0×10^{-7}/YR

FIRE AND FLOOD CORE DAMAGE FREQUENCY: 8.0×10^{-8}/YR

TOTAL CORE DAMAGE FREQUENCY (POINT ESTIMATE): 5.2×10^{-6}/YR

GESSAR PRA RESULTS SUBSTANTIALLY LESS THAN INTERIM SAFETY GOAL

GESSAR PRA

PRINCIPAL INSIGHTS

- GESSAR DETERMINISTIC REVIEW SHOWED THAT THE DESIGN IS TOLERANT OF SEVERE ACCIDENTS
- TRANSIENT INITIATORS MORE IMPORTANT THAN LOCAs
  - DOMINANT SEQUENCE - LOSS OF OFFSITE POWER (STATION BLACKOUT)
  - DESIGN CHANGES LIMITED ATWS IMPACT
- EXTERNAL EVENTS IMPACT GESSAR LESS THAN RECENT PLANT SPECIFIC PRAs
- REALISTIC EVALUATION OF FISSION PRODUCT TRANSPORT IS SIGNIFICANT
- EMERGENCY PROCEDURE GUIDELINES REDUCE THE LIKELIHOOD OF HUMAN FAILURE
- DATA BASE FOR INTERNAL EVENTS MORE COMPLETE THAN FOR EXTERNAL EVENTS
  - AGREEMENT USING MATURE METHODOLOGIES CONFIRMS THIS

GESSAR

PRA USE IN DESIGN PROCESS

- PRA RESULTS IDENTIFY AREAS OF POTENTIAL DESIGN IMPROVEMENT

  NRC PROPOSED 73 DESIGN IMPROVEMENT EVALUATIONS
  - GESSAR DESIGN ALREADY INCLUDED 23

  NONE OF THE REMAINING 50 FEATURES FOUND TO BE COST BENEFICIAL
  - RATIO OF COST TO BENEFIT GREATER THAN 1

- HYDROGEN CONTROL SYSTEMS NOT COST BENEFICIAL

  BY ORDERS OF MAGNITUDE

- SOME DESIGN IMPROVEMENTS CAN ADDRESS MORE FREQUENT TRANSIENT INITIATORS

  COMBINED IN PROPOSED ULTIMATE PLANT PROTECTION SYSTEM

  INCLUDES ADDITONAL VENTING, DEPRESSURIZATION, AND LOW PRESSURE WATER DELIVERY CAPABILITY

GESSAR

SEVERE ACCIDENT PROGRAM

CONCLUSIONS

- GE HAS MET ALL REQUIREMENTS OF SEVERE ACCIDENT POLICY
  - DESIGN SUBJECTED TO EXTENSIVE NRC REVIEW
  - SIGNIFICANT DESIGN CHANGES INCORPORATED
  - COMPREHENSIVE PRA PROVIDED

- THE GESSAR PRA HAS BEEN USED AS AN EXTENSION OF THE DETERMINISTIC EVALUATION
  - NUMEROUS DESIGN CHANGES WERE CONSIDERED

- THE PRA CONFIRMS THE SAFETY ACCEPTABILITY OF GESSAR
  - PROBABILITY OF CORE DAMAGE LESS THAN WASH 1400 AND INTERIM SAFETY GOAL
  - PLANT RISK WELL BELOW WASH 1400 AND INTERIM SAFETY GOAL
    - o NO CALCULATED EARLY FATALITIES

- THE NRC SAFETY REVIEW OF THE GESSAR DESIGN SHOWS THAT IT SATISFIES THE OBJECTIVES OF THE SEVERE ACCIDENT POLICY

BRIEFING FOR THE ACRS

ON

ACTIVITIES OF THE OFFICE FOR ANALYSIS AND EVALUATION OF OPERATIONAL DATA (AEOD)

MARCH 7, 1985

C. J. HELTEMES, JR., DIRECTOR
THOMAS IPPOLITO, DEPUTY DIRECTOR

A-230

APPENDIX XXIII
AEOD ACTIVITIES BRIEFING

SCOPE OF PRESENTATION

- AFOD ORGANIZATION
- PROGRAM SUPPORT ACTIVITIES
- TRENDS AND PATTERNS PROGRAM ACTIVITIES
- SUMMARY OF TECHNICAL STUDIES
- OPERATIONAL DATA INDICATORS OF PERFORMANCE

OFFICE FOR ANALYSIS AND EVALUATION OF OPERATIONAL DATA

A-232

Director
C. HELTEMES
Secretary G. THOMPSON

Deputy Director
T. IPPOLITO
Secretary C. GALLAGHER*

PROGRAM TECHNOLOGY BRANCH

Chief F. HEBDON
Program Assistant R. BECKER
Secretary E. CUNNINGHAM
Secretary R. BAKER

PROGRAM DEVELOPMENT SECTION

Chief R. DENNIG

L. BELL
M. HARPER
B. BRADY
K. HIGGINS
T. WOLF
P. O'REILLY

DATA MANAGEMENT SECTION

Chief J. CROOKS

P. BOBE
E. BOYLE
S. MASSARO
P. CROSS-PRATHER

REACTOR OPERATIONS ANALYSIS BRANCH

Chief K. SEYFRIT
Secretary J. ZECH
Secretary A. SMITH

REACTOR SYSTEMS SECTION 1

Chief S. RUBIN

T. CINTULA
P. LAM
S. SALAH
E. LEEDS

REACTOR SYSTEMS SECTION 2

Chief W. LANNING

H. ORNSTEIN
R. FREEMAN
D. ZUKOR
R. TRIPATHI

ENGINEERING SECTION

Chief M. CHIRAMAL

E. BROWN
F. ASHE
C. HSU

NONREACTOR ASSESSMENT STAFF

Chief K. BLACK
Secretary E. HOPPER

S. PETTIJOHN
E. TRAGER

*EOD 3/18/85

OFFICE FOR ANALYSIS AND EVALUATION OF OPERATIONAL DATA

STATUS OF SELECTED REACTOR PROGRAM ACTIVITIES

- LER REPORTING
  - RULE EFFECTIVE JANUARY 1, 1984
  - ISSUED SUPPLEMENT 1 TO NUREG-1022
  - STUDIES IN PROGRESS ASSESSING 1984 LERs
- DATA MANAGEMENT ACTIVITIES
  - SEQUENCE CODING AND SEARCH SYSTEM
  - PART 21/50.55E FILE
  - FOREIGN EVENT FILE
  - NUCLEAR PLANT RELIABILITY DATA SYSTEM (NPRDS)
- ABNORMAL OCCURRENCE REPORTING GUIDANCE
  - ESTABLISHED CRITERIA ON MEDICAL MISADMINISTRATIONS
  - REVISED NRC MANUAL CHAPTER 0212
  - ISSUED DETAILED STAFF GUIDANCE
- SPECIAL STUDIES
  - BNL STUDY ON NEED FOR INDEPENDENT SAFETY ORGANIZATION
  - SAMPLING OF LICENSEES' OPERATING EXPERIENCE ASSESSMENT PROGRAMS
- OPERATING EXPERIENCE FEEDBACK
  - ISSUE LER COMPILATION REPORTS (MONTHLY)
  - ISSUE POWER REACTOR EVENTS (PRE) (BIMONTHLY)
  - ISSUE AO REPORTS (QUARTERLY)
  - PARTICIPATE IN NEA-IRS ACTIVITIES

A-223

OFFICE FOR ANALYSIS AND EVALUATION OF OPERATIONAL DATA

TRENDS AND PATTERNS PROGRAM ACTIVITIES

- TASKS COMPLETED IN PAST YEAR
  - DEVELOPMENT OF TRENDS AND PATTERNS METHODOLOGY
    - CONTING PROGRAM GUIDE (NUREG/CR-3824 - SEPTEMBER 1984)
  - ANALYSIS OF UNPLANNED REACTOR SCRAMS (JANUARY-MARCH 1984)
    - AEOD/P406 - NOVEMBER 1984
  - FEEDWATER TRANSIENTS IN W PWRS (JANUARY 1981-JUNE 1983)
    - AEOD/P405 - DECEMBER 1984
  - ANALYSIS OF 1981 LERS
    - NUREG/CR-4071 - MARCH 1985
- TASKS NEARING COMPLETION
  - ANALYSIS OF 1981-1983 LERS
    - DRAFT NUREG/CR-4129 - JANUARY 1985
  - ANALYSIS OF UNPLANNED REACTOR SCRAMS (JANUARY-DECEMBER 1984)
  - ANALYSIS OF UNPLANNED ESF ACTUATIONS (JANUARY-JUNE 1984)
- TASKS UNDER DEVELOPMENT
  - ANALYSIS OF UNPLANNED SAFETY SYSTEM UNAVAILABILITY (1984)
  - ANALYSIS OF TECHNICAL SPECIFICATION VIOLATIONS (1984)
  - ANALYSIS OF CORRECTIVE ACTIONS
  - NPRDS - METHODOLOGY TRIAL USE

A-239

OFFICE FOR ANALYSIS AND EVALUATION OF OPERATIONAL DATA

STATUS OF TECHNICAL STUDIES

DOCUMENTS ISSUED JANUARY 1 THROUGH DECEMBER 31, 1984:

- o CASE STUDIES: 5 ISSUED } 25%
- o ENGINEERING EVALUATIONS: 31 ISSUED } 60%
- o TECHNICAL REVIEWS: 24 ISSUED } 15%
- o SPECIAL STUDIES: 5 ISSUED
- o PROGRAM SUPPORT STUDIES: 6 ISSUED

A-235

DOCUMENTS ISSUED IN PRIOR YEARS:

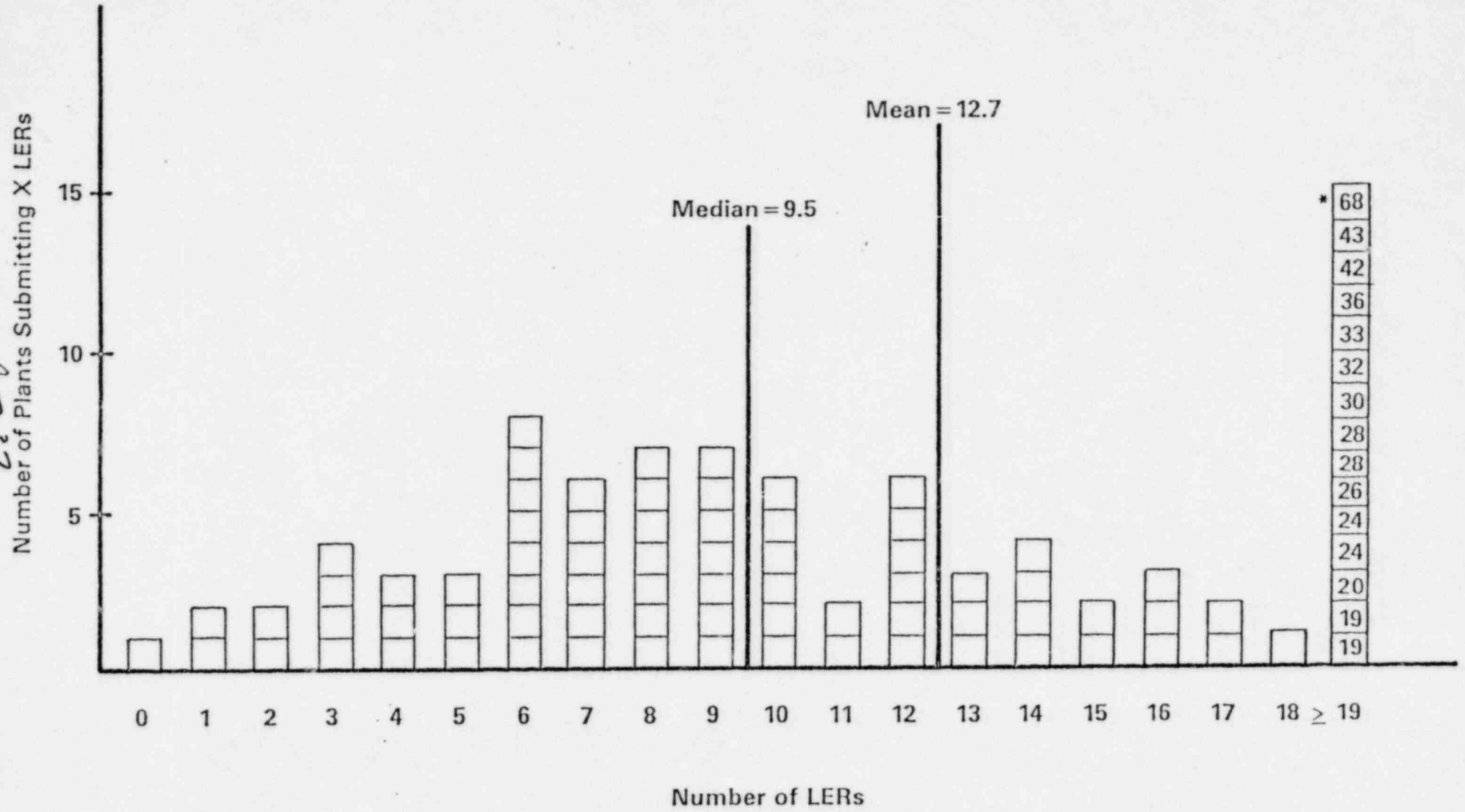
	1980	1981	1982	1983	1984
o CASE STUDIES	5	5	6	2	5
o ENGINEERING EVALUATIONS AND TECHNICAL REVIEWS:	20	41	66	82	55
o SPECIAL STUDY AND PROGRAM SUPPORT REPORTS:	0	0	0	1	11

POTENTIAL OPERATIONAL EXPERIENCE PERFORMANCE INDICATORS

- OPERATIONAL EXPERIENCE SHOULD BE A VALID AND DIRECT MEASURE OF PERFORMANCE
- DIFFICULTY IS IN SELECTING VALID, OBJECTIVE INDICATORS
- NUMBER OF LERs ONLY MAY BE AN INACCURATE AND MISLEADING INDICATOR
  - NOT ALL REPORTS HAVE SAME SIGNIFICANCE
  - FACTORS MAY INFLUENCE CONSISTENCY OF REPORTING, E.G., AGE, LOCATION, ATTITUDES, SIZE
  - TENDS TO MOTIVATE LICENSEES TO MINIMIZE THE NUMBER AND CONTENT OF LERs
- CRITERIA FOR VALID, OBJECTIVE INDICATORS FROM OPERATIONAL DATA
  - DATA MUST BE CONSISTENTLY REPORTED BY ALL LICENSEES
    - UNIFORM REPORTING CRITERIA
    - LITTLE OR NO JUDGMENT NEEDED TO IMPLEMENT
    - RESULTS CAN BE COMPARED DIRECTLY TO OTHER FACILITIES
  - CONDITION MUST BE EITHER DESIRABLE OR UNDESIRABLE
    - CHANGES IN RATE OF OCCURRENCE ARE EITHER POSITIVE OR NEGATIVE

A-236

Experience With LER Rule
LERs Submitted With Events Dates 1/1/84 Through 6/30/84



*Actual number submitted

POTENTIAL OPERATIONAL EXPERIENCE PERFORMANCE INDICATORS, CONTD.

- PROPOSED INDICATORS FROM LER DATA
  - RPS ACTUATION
  - OTHER ESF ACTUATIONS
  - SYSTEM UNAVAILABILITY
  - CONDITION PROHIBITED BY TECH SPECS
  - SHUTDOWN REQUIRED BY TECH SPECS
  - AIRBORNE AND LIQUID RELEASES
  - VOLUNTARY REPORTS

A-23 of

ATTACHMENT F

REPORTS TO THE ACRS ON THE
DOCUMENTS REVIEWED BY THE
WASTE MANAGEMENT SUBCOMMITTEE
DURING ITS MEETING ON
FEBRUARY 15-16, 1985

Report to the ACRS
on

10 CFR 60, Disposal of High-Level Radioactive Waste
in Geologic Repositories:
Amendments to Licensing Procedures

by the
Waste Management Subcommittee
February 15-16, 1985

General Comments:

1. Several members of the Subcommittee misinterpreted the proposed rule to imply that NRC would not issue a Site Characterization Analysis. Since others who read the proposed rule may make the same misinterpretation and since such a step would clearly be contrary to providing the public a range of opportunities for input into the review process, it might be wise to rewrite the proposed rule to reduce the possibilities for such a misinterpretation. Such a rewrite should make clear that it is only the draft Site Characterization Analysis that will not be issued.

2. Although the NRC recognizes the need for funding educational programs for, and technical and licensing evaluations by, State and local organizations, it should be acknowledged that most such support will have to be provided by the U.S. Department of Energy (DOE).

3. As in the past, the Subcommittee wants to emphasize that, during both the prelicensing and licensing stages, close liaison and interaction between the NRC and DOE waste management staffs is essential.

Specific Comments:

1. Page 15, first sentence, 1st full paragraph - Once again this demonstrates the need for close NRC-DOE interaction. Same paragraph, last sentence - the statement is made here that the site characterization analysis will be a dynamic process, suited for ongoing public input and NRC review. This appears contrary to statements by DOE staff members at recent public hearings[1], and contrary to the plans of the NRC to issue the SCA only in final form.

2. Page 29, Commissioner Asselstine's Additional Views:

   a. The Subcommittee believes that the NRC should monitor DOE's application of the guidelines, but should not independently evaluate the relative merits of the sites.

   b. In view of the many existing opportunities for public input into the review process, the Subcommittee does not believe that issuance of a SCA in draft form is necessary.

3. Page 32, paragraph 60.17, part (a)(3) - This leaves open the question as to what restoration is required for a site which, although suitable, was not selected for use as a repository. As a second comment, the Subcommittee believes that the phrase "site

restoration" is a more accurate description than "decontamination and decommissioning."

4. Page 32, paragraph 60.17 - It would be helpful to note that the site characterization requirements, as stated in the "Standard Format and Content Guide," still pertain and remain the same.

References:

1. Transcript of Proceedings Before the Department of Energy and Nuclear Regulatory Commission, Ref: "Commission's Comments on DOE's Mission Plan (for Civilian High-Level Waste Repository)," Washington, DC, dated December 17, 1985.

Report to the
ACRS on the

NRC Review Plan: Quality
Assurance Programs for Site
Characterization of High Level
Waste Repositories

Prepared by the
Waste Management Subcommittee
February 15-16, 1985

General Comments:

1. As a basic approach, the NRC has applied to HLW repositories the existing Quality Assurance Criteria for nuclear power plants. Although this may be useful, it generates problems because of differences between power plants and repositories. Some portions of the existing criteria do not appear to be applicable for site characterization; for one thing, the existing power plant criteria are not data-oriented.

2. It would be helpful if the plan could mention other key factors in which quality assurance is essential to the proper functioning of a repository: for example, the adequacy of the design, the appropriateness of the data being generated, the accuracy of interpretation of the data, and the characteristics of the host site. It is probably impossible to designate specific, generally applicable procedures to assure that geological and geophysical work, for example, is of high quality; thus, a proper approach must

include critical peer review to search for errors or omissions. Another key factor is management's commitment to a good QA program.

3. Since other Federal organizations, such as the U.S. Environmental Protection Agency, have developed extensive QA programs for various environmental assessments, it might be helpful for the NRC Staff to try to benefit from such efforts. We recommend that the NRC Staff check with EPA officials in this regard.

4. The Plan emphasizes the need for auditing the QA program, but insufficient attention appears to have been paid to the qualifications and experience requirements of the auditors. More emphasis with details would be helpful.

5. On March 21, 1984, the ACRS submitted to the NRC Chairman a report on the Draft NRC Report to Congress on improving quality and quality assurance in the design and contruction of nuclear power plants. Portions of that letter (see copy attached) that appear applicable here include:

    a. The Committee suggested that distinction needs to be made among quality, QA and QC and their relationship to safety. The same suggestion applies to the repository QA program. QA and quality are treated as if they are

identical (pg. 8, last paragraph); better definitions need to be included for both terms.

b. Another portions of the Committee's QA letter that would carry over to repositories is the suggestion that PRA be used for identifying systems and components for which quality is essential to public safety. The repository QA program document is silent on this subject.

6. The Plan, as published, appears to be in final form. Since it is undergoing review and changes are anticipated, it might have been better to label it as a "draft".

7. The Plan describes the need for close liaison between the NRC and DOE waste management staffs, but the impression is often given that such interactions are difficult to implement. Although perhaps outside the scope of the QA Plan, the Subcommittee urges that steps be taken to assure the continuing and free flow of information between the two agencies, including regular and frequent meetings of corresponding staff members.

Specific Comments: Main Text

1. Page 5, Section 1.0 - it would be helpful if the opening paragraph

could refer to the fact that these comments and recommendations pertain to a HLW repository.

2. Page 6, Section 2.1, third paragraph - the definition of an accident is not clear. Over what time period is the 0.5 rem dose assumed to occur? And is the reader to understand that, after permanent closure, a different definition will apply?

3. Page 7, underlined sentence, middle of the page - what are "appropriate quality programs"?

4. Page 7, last paragraph - why must the QA Plan state that the "NRC Staff shall be permitted to visit and inspect the site ..."? Presumably they have this right as part of their prelicensing responsibilities.

5. Page 8, last paragraph - the definition of "quality assurance" given here is not correct. QA, alone, cannot assure that subsystems and components will perform satisfactorily. Consideration might be given to adding a footnote here, or elsewhere, stressing that QA is but one part of an overall program needed to assure proper operation of a HLW repository. Also needed are proper interpretation of data, good design, management dedication, peer review, etc. QA is primarily concerned with the conformance of the facility with the Technical Specifications.

6. Page 9, first line - QA also must include an examination of the data, including how they were obtained.

7. Page 12, second paragraph, lines 8 through 10 - this sentence needs editing. Site visits by the NRC Staff will be required regardless of whether the procedures are made a part of the SCPs.

Specific Comments: Appendix A

1. Page 1, paragraph 1.4 - here again, the qualifications and training necessary for the auditors should be addressed.

2. Page 7, paragraph 3.8 - a peer review should be required for old procedures as well as new.

3. Page 7, paragraph 4.1, line 7 - Procurement documents must (not should) require....

4. Pages 11 and 12 - also to be addressed here is the quality of the data taken from the existing literature. The National Bureau of Standards has a procedure for judging the quality of data that might be investigated.

5. Page 12, paragraph 11.2, Sections 3.7, 3.8 and 3.9 - these sections do not appear to apply directly to a repository.

6. Page 12, Section 11.3, bottom of the page - this should be numbered Section 11.5.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

March 21, 1984

The Honorable Nunzio J. Palladino
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Dr. Palladino:

SUBJECT: ACRS REPORT ON DRAFT NRC REPORT TO CONGRESS ON IMPROVING QUALITY AND THE ASSURANCE OF QUALITY IN THE DESIGN AND CONSTRUCTION OF COMMERCIAL NUCLEAR POWER PLANTS

During its 287th meeting, March 15-17, 1984, the Advisory Committee on Reactor Safeguards reviewed the draft NRC report to Congress, "Improving Quality and the Assurance of Quality in the Design and Construction of Commercial Nuclear Power Plants," Revision 3, dated March 13, 1984. An earlier version of the draft report was considered during a meeting of the ACRS Subcommittee on Quality and Quality Assurance in Design and Construction held in Washington, D.C. on February 24, 1984. In addition, the Subcommittee reviewed and discussed the NRC's quality-related initiatives during meetings held on July 18, 1983 and December 6, 1983.

The report is both useful and constructive. It gives thoughtful attention to the five alternatives which the Commission was required to consider under Section 13(b) of the Ford Amendment (Public Law 97-415) and reaches well-reasoned conclusions on each. Further, the results of the pilot program mandated under Section 13(c) of the Ford Amendment substantiate the conclusion that comprehensive audits of nuclear construction projects by qualified third parties can provide significant additional preventive and detection capability as well as enhanced assurance that nuclear plants are built in accordance with their design and licensing commitments. The report is candid in conceding errors of omission or commission on the part of the NRC which have contributed to quality assurance deficiencies in the past.

During the Subcommittee's early review of the study, it suggested that the Commission take advantage of the opportunity presented by Congress and expand the scope of the study to address issues beyond those mandated. We are pleased that the report provides a more comprehensive picture of the Commission's actions and initiatives.

Although the report is well written, it is voluminous and repetitious. A concise executive summary would improve the report. This can be accomplished without delaying the submission of the report to Congress.

The lessons learned from past problems in the design and construction of commercial nuclear power plants are described. As indicated in the report, little is said about the operation of plants although many of

A-249

the same observations and lessons should apply. Many of the problems described relate to the inability, or the difficulty, of assuring the quality of some plants as a result of shortcomings in quality assurance and/or quality control programs during design and construction. Not addressed is whether the QA/QC shortcomings had an effect on quality or had significant effect on public safety/risk.

The distinctions among quality, quality assurance and quality control, and their relationship to public safety/risk are, at times, not made clear in the report. This is compounded by the NRC's continued inability to clearly identify those systems and components for which quality is essential to public safety and thus for which programs to control and to assure quality are necessary. Probabilistic risk assessments (PRAs) could help in this regard. We encourage the NRC to expedite the collection of the data necessary to clarify these issues.

Further, although recommendations are based on the findings of the case studies, pilot programs, and other initiatives, it is not clear whether their implementation will actually improve quality or enhance public safety or whether they will merely improve the public's perception of safety. We recommend that the NRC Staff undertake to determine the relative risk significance of the various recommendations and proposed actions as well as determine whether safety would be enhanced by the proposed actions. The NRC should then concentrate its efforts on actions which will enhance public safety.

The report does not contain priorities or schedules for further development of the various recommendations or proposed actions. We believe that the NRC Staff needs to develop more specific recommendations following the submission of the report to Congress. We recommend that in forwarding this report to Congress, the Commission make clear it intention to develop a plan for achieving the assurance of quality in the design, construction, and operation of nuclear power plants. However, we caution that the development of a program plan should not be allowed to interfere with proceeding expeditiously with those actions found to improve public safety significantly.

The NRC Staff has identified management as a major factor affecting the success or failure in assuring the quality and safety of nuclear power plants. While we agree that a poor quality assurance program is an indication of poor management, an apparently good quality assurance program does not necessarily imply the presence of good management. We see the need for an organizationally independent quality assurance department that reports to senior management; however, we fear that the emphasis on independence has in some cases led to the belief that the assurance of quality is someone else's responsibility. To assure quality and public safety, a strong sense of the need for and the benefits from quality, the assurance of quality, and professionalism should permeate a licensee's and/or a vendor's entire organization. The

NRC should continue its efforts to stimulate this kind of professionalism in the nuclear industry.

One of the recommendations from the management analysis conducted by N. C. Kist & Associates, Inc. is to establish a quality assurance program within the NRC. Although noted in passing in the report, it remains a fallow recommendation. The report does contain a recommendation for performance audits of NRC QA activities. However, we believe that the relationship between QA and prudent management, as discussed in Section 3.5 of the report and in this letter above, is equally applicable to the entire NRC. Therefore, we suggest that the Commission give prompt and careful consideration to the recommendation that the NRC establish a program to assure the quality of its activities. We do not believe, however, that a formal QA program is necessary or desirable.

The recommendation that the NRC establish a body of experts to advise the Commission on the capability of applicants to effectively manage a nuclear construction project is worthy of further consideration. The ACRS currently does not contain extensive expertise of the types envisioned for the proposed advisory body, and to establish such expertise within the ACRS membership might sacrifice other requisite expertise. The report has also recommended that future construction permits be conditioned on a demonstration of the licensee's continuing ability to effectively manage the project. Those responsible for the development of these recommendations should consider the difficulties associated with judging such management capabilities.

The ACRS supports the NRC Staff's shift in inspection emphasis from looking at the content of quality assurance plans to looking at actual plant quality and at the implementation and effectiveness of programs to assure quality. However, we believe that the NRC Staff will experience difficulties implementing the modified inspection program until performance criteria are established.

Useful insights have been obtained from Integrated Design Inspections (IDIs), Independent Design Verification Programs (IDVPs), and Construction Appraisal Team (CAT) inspections. We recommend that for the present these inspection programs continue.

The concept of using designated representatives is worthy of further consideration. In addition to augmenting NRC resources, it may be a way of stimulating and rewarding professionalism and dedication to quality in the workplace. We would like to be kept apprised of the NRC Staff's efforts regarding the designated representative concept and other QA initiatives, and we would appreciate the opportunity to comment on them at a later date.

A-258

Additional comments by ACRS Member Glenn A. Reed are presented below.

Sincerely,

Jesse C. Ebersole
Chairman

Additional Comments by ACRS Member Glenn A. Reed

I consider the report to Congress to be deficient in its study of Alternative b(4), and I am concerned that Congress may continue endorsing regulatory approaches that are too similar in many ways to those that have in the past proved ineffective. I concur with this ACRS report concerning the above referenced report to Congress in most aspects, but in my opinion it does not go far enough, and is not critical enough with respect to the following:

1. The ACRS report states that the report to Congress "gives thoughtful attention to the five alternatives ...." I disagree that thoughtful or appropriate in-depth attention was given to Alternative b(4), which addresses improvements in the NRC's organization, methods, and programs for quality assurance.

2. The report to Congress recommends that a body of experts be established to advise the Commission on an applicant's management capabilities. The ACRS report states that this recommendation is worthy of further consideration. I disagree, and do not feel such expertise, with the time and objectivity, could be constituted to undertake this activity. Further, I disagree that such a body of experts is even desirable or necessary if a more astute study of Alternative b(4) is made.

3 What the ACRS report does not address, or recommend, is more in-depth consideration of the NRC's organizational structure, and what obstacles this present structure may place in the path of achieving quality in design and construction. The present NRC structure does not motivate professionalism and craftsmanship in the workplace. In my opinion, high quality can only be achieved by the enthusiastic and dedicated action of real professionals and crafts people who are motivated to standards of excellence by a regulatory structure that better recognizes human factors. I am aware of and have read an NRC Staff report which addresses the FAA designated representative (DR) system. In my opinion, the report to Congress should not have glossed over the FAA's DR program, but should have included a detailed study of that system and its potential for correcting the adversarial climate that is growing in the nuclear workplace. Given the current structures of the nuclear

A-32

industry and the NRC, the genuine professionals and crafts people are somewhat overwhelmed by top brass and regulations, yet the answers for real quality in this highly technical nuclear industry lie with those professionals.

In my opinion, the achievement of a high degree of design and construction quality can come from a modified version of the FAA system of DRs in design and architect-engineer organizations and in manufacturer and constructor shops. I would consider it appropriate for these DRs to be nominated by their peers, approved by their employers and perhaps the NRC, then established in a quasi-regulatory role while continuing their regular duties. Along similar lines, the NRC might consider structuring some licensed personnel in nuclear power plants into a DR system somewhat similar to the way in which the Massachusetts Department of Public Safety has incorporated licensed operating engineers into its regulatory structure.

References:

1. Public Law 97-415, NRC Authorization Act for Fiscal Years 1982 and 1983, Section 13 on Quality Assurance, dated January 4, 1983.

2. Draft NRC report to Congress, "Improving Quality and the Assurance of Quality in the Design and Construction of Commercial Nuclear Power Plants," Revision 3, dated March 13, 1984.

Report to the ACRS
on the

"Final Technical Position on
Documentation of Computer Codes
for High-Level Waste Management,"
NUREG-0856

prepared by the

Waste Management Subcommittee
February 15-16, 1985

General Comment:

1. Organizations involved in the use of computer codes in environmental assessments often do not have good control of the codes they use. If performance assessments are to be required, strong in-house control will be necessary.

Specific Comments:

1. Page 1, third paragraph, line 6 - This calls for "theoretical documentation" of the codes. What is probably needed is the "documentation of the theory" used in the codes.

2. Page 4, item (3) - It would be helpful to describe the specific inputs that are needed and the specific outputs that are anticipated.

3. Page 4, Section B - In addition to a derivation and justification of the model, some evaluation should be made whether it is a valid model and whether it applies to the given situation.

4. Page 10, Section (3), middle of the page - This section appears to be in error. It will be necessary to validate each modification of a code.

REPORT TO THE ACRS
ON THE
DRAFT TECHNICAL POSITION SUBTASK 1.1:
WASTE PACKAGE PERFORMANCE AFTER REPOSITORY CLOSURE,
NUREG/CR-3219, VOL. 1

by
ACRS Waste Management Subcommittee
February 15-16, 1985

General Comments:

1. The purpose of this report is not clear to the Subcommittee. The report appears to be primarily directed to the more difficult problems likely to be encountered in evaluating the performance of a waste package after repository closure. As a result, it primarily describes what cannot be done.

2. In these evaluations, major attention has been directed to assessments of the waste form and its container; little attention has been directed to the backfill and only titanium containers have been assessed. Neither stainless steel nor associated stress corrosion cracking was considered.

3. The Subcommittee is aware that the report was prepared over two and one-half years ago. During this time, many newer data have been developed. Since the NRC Staff indicated that this work is ongoing, the Subcommittee looks forward to receipt of any subsequent reports that may be issued.

4. The list of references in the existing report is very limited, and those that are listed are based primarily on work done at the

Brookhaven National Laboratory. A more comprehensive use of references would have been helpful.

Specific Comments:

1. Page iii, Abstract - The first sentence should clearly state that the development of this report was sponsored by the NRC. It might also point out that the report was published without internal NRC review, if this is the case.

2. Page 12, Section 5.3 - We do not understand the first sentence. Why is a "validated isothermal rate expression" the "only rigorous means of predicting long term behavior"?

3. Page 12, Section 6.1 - The meaning of the first sentence is not clear. How can a null test require "a reproducible demonstration of non-detectable release of radionuclides...."?

4. Page 14, Section 6.5 - How can tests be "primarily qualitative in nature and design specific"?

5. Page 23, Section 10.1 - This portion is good as far as it goes, but why restrict the discussion to titanium?

6. Page 25, footnote - Unpublished manuscripts are not readily available.

7. Page 27, Section 10.3.4 - This covers pH and flow-rate effects. Where are redox (Eh) measurements discussed?

8. Page 34, Section 10.6, item (2) - "Hard rock" usually refers to igneous or metamorphic rock. It is not clear which is meant here.

Report to the ACRS
on the

"Draft Technical Position -
Subtask 1.2 Post Emplacement
Monitoring," NUREG/CR-3219, Vol. 2

prepared by the
Waste Management Subcommittee
February 15-16, 1985

General Comments:

1. As written, the report appears to be designed to assure that every possible parameter is monitored at a maximum number of locations. Some attention might be given to assessing which parameters are the most important, which are necessary, and whether the number of monitoring stations might be reduced.

2. The report appears to treat the monitoring of a repository in the same manner as a nuclear power plant. In the nuclear plant, unusual events can occur rapidly; in a repository, they may require months, years, decades or more to take place. As a result, the Subcommittee questions the need for a repository monitoring system to have uninterrupted power supplies, system redundancies, and immediate alarms. A basic change in philosophy may be in order.

3. In terms of philosophy, some attention might also be given to the consideration of the installation of self-contained retrievable monitoring systems that could be withdrawn for examination, repair and/or replacement.

4. Although the Subcommittee concurs that random location of monitors within the repository is useful in assuring good statistics, some attention should be given to locating extra monitors in those areas considered to be most vulnerable. The development of possible unusual event scenarios, with accompanying potential impacts, might be helpful in determining where best to locate monitoring systems and the most critical parameters to be measured.

Specific Comments:

1. Page 1, second paragraph, line 7 - This sentence needs editing; one possible approach would be to say: "Stresses on the waste package will come from both lithostatic and tectonic phenomena, the latter of which ....."

2. Page 3, last paragraph, second line - We would suggest you say: "...shown that instruments will almost certainly have to be altered...."

3. Page 4, Section 2 - the report appears to address all likely repository media except for tuff in locations above the water table.

4. Page 9, line 3 of text in the middle of the page - Most salt deposits do not include sylvite (KCl).

5. Page 16, Section 2.4 - The time periods cited here appear to be low. The statements on page 28, last paragraph, appear to reflect a more accurate description of the situation.

6. Page 18, last line - The "G-Tunnel" at the NTS is not located in tuff. The same error is made on page B-2.

7. Page 19, Section 2.5 - This might read better if the first word were "Pressures." Then we would suggest you add a third sentence: "Differential, horizontal and vertical stresses would also be reestablished over time."

8. Page 19, Section 2.5.1 - We would suggest the last two sentences of the first paragraph be reworded as follows:

   "In addition to the hydrostatic and lithostatic pressure on the waste package, significant differential stresses are observed. At repository depths, horizontal stresses can be ....."

9. Page 27, Section 3.2.1.4, second paragraph, line 13 - "hard rock" usually refers to metamorphic or igneous rock. It is not clear which is meant here.

10. Page 27, last paragraph, lines 6 through 10 - This implies that the monitoring will be done only while the drifts are open. Will this

not severely restrict the post emplacement monitoring period? The same comment applies to page 34 (last paragraph).

11. Page 33, last full paragraph - If breaching of waste package components is not permitted, will any monitoring be done on conditions within the packages? On page 26, the statement on "Mechanical property" (middle of the page) implies that such monitoring will be necessary.

12. Page 38, middle paragraph - Is it anticipated that the hydrogen will remain in the vicinity of the waste package? What is to prevent it escaping? If it does, then you would not have a hydrogen rich atmosphere; in fact it might be an oxidizing environment.

13. Page 44, Section 5.2, second bullet - What are "pH excursions"? Are there time and pH changes that apply?

14. Page B-1 - This is a useful discussion but we question whether it is "state-of-the-art." The section on instrumentation that has proven useful in field monitoring to date is, however, very informative.

15. Page B-23, first paragraph - Measurements of pH under laboratory conditions will have no meaning since the CO_2 will have been lost, etc. The ideas are here but the paragraph needs to be rewritten.

16. Page B-23, second paragraph - The discussion of the Eh is out-of-date; newer information is available.

ADDITIONAL DOCUMENTS PROVIDED FOR ACRS' USE

1. Handout during AEOD Activities session at ACRS full Committee meeting March 7, 1985, AEOD Studies Issued Since September 21, 1984 Semiannual Report.