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JAN 9 1985

# MEETING MINUTES FOR THE
# ACRS HOPE CREEK GENERATING STATION UNIT 1 SUBCOMMITTEE
# NOVEMBER 28-29, 1984 - PHILADELPHIA, PA.

The ACRS Subcommittee on Hope Creek Generating Station Unit 1 held a meeting on November 28-29, 1984, at the Sheraton University City Hotel, 36th & Chestnut Streets, Philadelphia, Penn. The purpose of the meeting was to review the application by Public Service Electric and Gas Company (PSE&G) for a license to operate Hope Creek Nuclear Generating Station Unit 1. As part of the review, the Subcommittee and its consultants toured the plant. The meeting was open to the public. Notice of this meeting was published in the Federal Register on November 9 and 21, 1984. A copy of this notice is included as Attachment A. The meeting schedule is Attachment B. A complete set of handouts and a list of attendees have been included in the ACRS files. There were no written or oral statements from the public. The Designated Federal Official for this meeting was Mr. Gary Quittschreiber.

Principle Attendees:

ACRS
C. Siess, Chairman
J. Ebersole, Mbr
C. Michelson, Mbr
M. Carbon, Mbr
P. Pomeroy, Con
G. Quittschreiber, Staff
M. El-Zeftawy, Staff

NRC
D. Wagner
A. Schwencer
P. Sobel
J. Chen
F. Allenspach
R. Starostecki, Region I

PSE&G
R. Eckert
R. Uderitz
T. Martin
B. Preston

DESIGNATED ORIGINAL

Certified By Bgk

W. Pavincich
W. Gailey
S. LaBruna
C. Churchman
E. Yochheim
J. Rogozenski
A. Giardino
P. Moeller
H. Hanson
R. Lovell
T. McGuire
P. Landrieu
R. Salvesen
Y. Yaworsky
J. Wroblewski
R. Douglas
C. Johnson
S. Kosierowski
W. Britz

Meeting Highlights

Dr. C. Siess opened the meeting. He called upon the NRC Staff for the first presentation.

I) NRC Staff Presentation:

D. Wagner, Project Manager-Division of Licensing, stated that on February 27, 1970, Public Service Electric and Gas Company filed with the Atomic Energy Commission (AEC) an application for a license to construct and operate the Newbold Island Nuclear Generating Station, Units 1 and 2. The site was located in the Township of Bordentown, Burlington County, New Jersey. By Amendment 13 (submitted November 1, 1973), the Newbold Island physical plant was relocated adjacent to the Salem Generating Station and renamed the Hope Creek Generating Station, Units 1 and 2. The Hope Creek facility is located on an Artificial Island in lower Alloways Creek Township, Salem County, New Jersey, approximately 18 miles southeast of Wilmington, Delaware.

The AEC reported the results of its Construction Permit (CP) review in an SER, dated December 17, 1971. Following a public hearing before an Atomic Safety and Licensing Board, Construction Permits were issued on November 4, 1974. In December of 1981, Unit 2 was cancelled.

Public Service Electric and Gas Company on behalf of itself and acting as agent for the Atlantic City Electric Company, filed a Final Safety Analysis Report as part of the OL application request, dated March 1, 1983.

The Applicant has retained Bechtel Power Corporation to provide architectural-engineering, construction, and start-up services.

The NRC Staff has published the results of its safety review of Public Service Electric and Gas Company's application for a license to operate the Hope Creek Generating Station Unit 1. The Hope Creek plant will use a boiling water reactor (BWR/4) with a General Electric Mark I containment. The nuclear reactor is similar to Limerick, Susquehanna, and Hatch. The design power level of the reactor is 3,435 MWt. The rater power level is 3,293 MWt. The net electical output is 1,067 MWe. The SER summarizes the results of the NRC Staff technical evaluation of the plant.

The NRC Staff has identified 15 open items that have not been resolved with the applicant, namely:

Item 1 - Riverborne Missiles - The probable maximum flood is due to the probable maximum hurricane which results in flooding plant grade to a depth 12.3 feet. Because the Delaware River is a navigable waterway, the applicant must address the effects of ships and boats with a draft of less than 12 ft. hitting the walls and penetrations (doors) of safety related structures.

Item 2 - Equipment Qualification - Generally, this is an open item because the in-depth equipment qualification review typically commences late in the review process.

Item 3 - Preservice Inspection Program - Staff review of the PSI plan indicates that the use of cladding on piping welds may interfere with ultrasonic examination.

Item 4 - GDC-51 - The Staff was unable to conclude, relative to fracture toughness, that a sufficient margin of safety existed under limiting environmental condition to be experienced by the feedwater check valves.

Item 5 - Solid State Logic Modules - The Hope Creek design incorporates the use of Bailey 862 Solid State Logic Modules. The Staff is reviewing the use of these modules with particular emphasis being placed on the common manual and automatic initiation capability.

Item 6 - Post Accident Monitoring Instrumentation - The Staff is reviewing the applicant's response to Regulatory Guide 1.97, Revision 2, to determine the adequacy of post accident monitoring instrumentation at Hope Creek.

Item 7 - Minimum Separation Between Non-Class 1E Conduit and Class 1E Cable Trays - The applicant has not demonstrated the acceptability of 1-inch minimum separation between steel conduit and 1E cables located in an open cable tray.

Item 8 - Control of Heavy Loads - The applicant has not submitted acceptable responses to NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants."

Item 9 - Alternate and Safe Shutdown - The Staff is still reviewing alternate and Safe Shutdown information which was submitted by the Applicant.

Item 10 - Delivery of Diesel Generator Fuel Oil and Lube Oil - The applicant should describe how fuel oil will be delivered to the site during flood conditions.

Item 11 - Filling Key Management Positions - The assignment of individuals to key management positions has not been completed.

Item 12 - Training Program Items - This is a categorized as an open item because the following areas are considered incomplete:

A. Initial Training Program

B. Requalification Training Program

C. TMI Items I.A.2.1, I.A.3.1, II.B.4

D. Replacement Training Programs

E. Nonlicensed Training Programs

Item 13 - Emergency Dose Assessment Computer Model - The applicant is developing an emergency dose assessment computer model for inclusion in the central radiation processor. The applicant must provide a description of this model.

Item 14 - Procedures Generation Package - The applicant has not submitted the emergency procedures generation package for staff review.

Item 15 - Human Factors Engineering - The applicant has not submitted the SPDS safety analysis, additionally, the Staff is currently reviewing the applicant's compliance with TMI item II.K.3.27 and the applicant's DCRDR Summary Report.

There are 37 confirmatory items (Attachment C) that have essentially been resolved to the Staff's satisfaction, but for which certain confirmatory information has not been provided by the applicant, and 7 items (Attachment D) for which a license condition may be desirable to ensure that Staff requirements are met by a specified date.

II) Applicant's Presentation:

A) Richard Eckert, Senior Vice President (PSE&G), stated that the prime objective of PSE&G company is to provide safe and reliable electric power at a reasonable cost to the customers. All decisions which could affect the health and safety of the public will be made conservatively.

B) William Gailey, Chief Project Engineer, summarized certain Hope Creek design features which are either unique or will enhance the safe operation of the plant. He stated that ultimate heat sink for the plant is the Delaware River which feeds the service water system. Waste heat from the condenser is dissipated by the circulating water system through the cooling tower. Mr. Gailey stated that:

- PSE&G personnel spent more than 20,000 man hours conducting design reviews utilizing the drywell model. Major concerns such as ALARA, studies on maintainability, and accessibility were addressed.

- In 1977, PSE&G conducted a control room operability review using a full scale control room mock-up. The purpose was to make sure from a human engineering standpoint that the latest industry experience including actual plant operations were being factored in.

- Hope Creek has a simulator which is identical to the control room.

- Hope Creek's secondary containment is a cylindrical reinforced concrete structure with a reinforced steel-lined dome. The purpose of this design is to provide as low a leakage structure as possible.

- A filtration, recirculation and ventilation system (FRVS) has been included to recirculate and filter the atmosphere in the reactor building. The FRVS is designed to collect air-borne contamination released to the reactor building and by mixing, filtering and maintaining negative pressure that would minimize radioactive releases.

Mr. Michelson asked if the reactor building consists only of the cylindrical portion or if the corner rooms are included and, if so, are the corner rooms included as a part of the FRVS.

Mr. Drewnowski, PSE&G, replied that the square section is also used as the reactor building and is serviced by the FRVS.

- In addition to the two main normal isolation stop valves, a third valve has been added to each main steam line to minimize leakage during shutdown. A seal air system was also added to further reduce leakage.

Dr. Siess mentioned that most of these changes were originally made for the Newbold Island plant to reduce off-site releases following a design basis accident. Changes to the high-pressure core spray and LPCI were in response to ACRS concerns at that time. Those changes were made to meet ACRS demands for improved ECCS and were all features put in during extensive review of a large reactor at a highly populated site. All of these changes were maintained after the plant was moved to Hope Creek site to expedite licensing.

Mr. Gailey went on to mention another design feature is high pressure coolant injection via the core spray sparger. In addition, the turbine-driven HPCI pump was redesigned to provide additional 12% of flow to the core. Another feature is the safety auxiliary cooling system which is a closed loop and only its heat exchangers are exposed to the Delaware River.

C) In addressing the subject of ATWS, the Applicant has incorporated the ATWS-3A modifications. This provides for automatic initiation of standby control system, tripping recirculation pumps, alternate rod insertion and feedwater runback.

D) Mr. P. Landrieu, PSE&G, Construction Manager, stated that the fuel load date is January 15, 1986; however, the target schedule is December 1, 1985. Construction is approximately 93 percent complete. Start-up status is 34 percent complete.

E) Mr. S. LaBruna, Assistant General Manager for Hope Creek, discussed the plans to handle maintenance, inservice inspection and pre-operational testing. A fundamental aspect of maintenance of PSE&G facilities during the past decade has been a repair and maintenance procedures systems (RAMPS), which defines requirements for a planned approach to maintenance activities using detailed procedures that results in quality and efficient use of personnel. The cornerstones of the managed maintenance program are the preventive-predictive-corrective maintenance and integrated spare parts and component data file.

Dr. Carbon asked if the Applicant is expecting to get any help from EPRI and other BWR operators? The Applicant replied, yes; certainly from the NPRDS standpoint some of the concerns have been already in the industry.

Mr. LaBruna discussed the inservice inspection program, stating that during plant operation, the nuclear service department will coordinate inservice inspection and surveillance activities and analyze test results in accordance with the 10 CFR 50 Appendix J.

The preoperational test program is an integrated effort between engineering construction and nuclear department personnel. The first stage of the preoperational test program includes instrument calibration, energized checkout of the electrical power and control system, piping flushes and isolated equipment operation. The second stage of the program demonstrates the capability of systems to satisfy design intent. All records generated during the testing program will be retained for the full life of the plant in accordance with Regulatory Guide 1.68.

F) Mr. Gailey, PSE&G, discussed resolution of comments from previous ACRS letters. Dr. Siess commented that there is no need to address anything in a letter that clearly was related to the Newbold Island site.

There was an ACRS Hope Creek letter in 1974, which contains three items of concern. The first item identified was a request to reevaluate core operating limits as a result of what was then recently promulgated acceptance criteria for ECCS.

Mr. Gailey stated that this reevaluation was done back in 1974 and, to be consistent with current Staff requirements, PSE&G is currently redoing the analysis.

The second item addressed in the 1974 ACRS letter related to foundation soils. Although the Hope Creek is adjacent to Salem and has essentially the same seismological, geological and foundation conditions, Hope Creek had committeed to undertake soil testing programs and a specific Hope Creek design would be reviewed by the NRC Staff. Mr. Gailey stated that these programs were undertaken and completed. They have been satisfactorily reviewed by the Staff with no open items.

The third item was a commitment on PSE&G's part to conduct a probability analysis of waterborne accidents that could affect plant safety. These are accidents on the Delaware River.
Mr. Gailey stated that this analysis was completed and indicated a very low probability of an accident occurring. The NRC Staff concluded that it is not a design basis accident.

Dr. Siess commented that thirteen years ago there was a paragraph in the Newbold Island ACRS letter on hydrogen control and recommendation that the containment should be inerted, which has been followed. In addition, to which PSE&G has recombiners now.

Dr. Siess asked if the Applicant has made any design provisions to reduce the quantity of reactor coolant discharged to the reactor building in the event of a process line break. This has to do with instrument lines that went through the drywell and had flow restricting orifices.

Mr. Gailey, PSE&G, replied as far as the instrument lines are concerned, the orifices were retained, in addition to excess flow check valves, concluding that the instrument lines are not a further concern. As far as the process line breaks are concerned, the intent was to provide some features to keep the off-site releases well within the 10 CFR 100 guidelines.

Mr. J. Ebersole, commented that the Applicant should look not only at the dose consequences but rather at the environmental impact on critical equipment and people (e.g., the effects of condensation on surfaces which lead to short circuits, etc.)

G) Seismic design of plant and equipment - Dr. Siess asked if the NRC Staff has treated the more recent information regarding SSE for New Brunswick earthquake (which was real) and the unleashing of the Charleston earthquake (which may or may not be real).

Phyllis Sobel, NRC Staff, said that the maximum historic events within about 200 miles of the Hope Creek site were of about epicentral intensity VII, and about maximum estimated magnitude 5. These events occurred in Asbury Park, New Jersey, which is on the northern coast of New Jersey, near New York City. The January 1982 New Brunswick earthquake, of magnitude 5.75, occurred in a cluster of seismicity at about latitude 47 degrees. Although about 700 miles from the Hope Creek site, this New Brunswick event is significant because it occurred in the New-England Piedmont tectonic province. The closest approach to the province is about 18 miles northwest of the site. The maximum events within about 50 miles of the site were about magnitude 4. So, the NRC Staff did not believe it was possible for a magnitude of 5.75 event to occur within about 200 miles of the Hope Creek site. To be conservative, the NRC Staff looked at the possibility of a magnitude 5.75 event occuring about 18 miles from the site. The Staff found that the site SSE was adequate for describing the ground motion effects of the New Brunswick earthquake at that distance.

For the 1886 Charleston earthquake, the Staff's position was presented to the ACRS in April of 1983. That position includes both deterministic and probabilistic studies. The deterministic study should reduce the uncertainty by better defining the causal mechanism of the Charleston earthquake. The probabilistic program is being done by Lawrence Livermore National Laboratories. It includes the use of expert panels for seismicity and ground motion inputs and sensitivity studies. The basic objective is to identify those sites that have a high hazard with respect to their design. The Hope Creek site is not one of the first ten test sites.

Dr. P. Pomeroy, ACRS Consultant, asked the NRC Staff if they have site-specific seismic hazard curves other than the ones generated by Lawrence Livermore Labs. The Staff replied no; however, Dr. Robin McGuire as a consultant for the Applicant replied that he undertook a study which involved replication of the assumptions used in the Lawrence Livermore study for those ten other sites.

Mr. Charles Churchman, Engineering Site Manager - PSE&G, discussed the seismic analysis, particularly in the area of soil-structure interaction and soil liquefaction. He stated that the liquefaction potential for Category I structures was determined by comparing the shear stresses induced in the soil by the SSE with the cyclic shear strength of the soil in the field condition. The maximum shear stresses at various points in the foundation were obtained from dynamic analysis. The Hope Creek Project has three major Category I foundation systems: power block, service water intake, and the service water pipeline.

Dr. Harry B. Seed, University of California at Berkeley, has been retained by the Applicant to evaluate the safety margins against liquefaction for the soils at the Hope Creek site. Dr. Seed mentioned that the Hope Creek site has a lot of relatively dense sands. The properties of these sands were measured by laboratory loading tests and because of its dense condition it has a very high liquefaction resistance. The approach chosen to evaluate the liquefaction resistance was to consider the boudnary line separating the liquefiable sites from nonliquefiable sites. The 7.5 magnitude earthquake was chosen and was extrapolated to smaller magnitude earthquake. Based on Dr. Seed's discussion, it was judged that the Category I foundations for the Hope Creek site are not only adequate for the design SSE of 0.2g, but also have sufficient seismic margin of safety. Dr. Seed defined the factor of safety as the number by which the acceleration could be increased higher than the postulated SSE value without bringing the soils to a condition of failure.

H) Mr. William Pavincich, Principal Engineer PSE&G, described the passive fire protection features designed into Hope Creek, namely:

- Separation of safe shutdown equipment by rated fire barriers or adequate distance.

- Use of IEEE 383 qualified cable.

- Use of UL approved building materials and components such as fire doors.

- Use of approved penetration seals for openings in rated fire barriers.

Also the active fire protection features:

- Automatic early detection systems.

- Multi-faceted automatic primary suppression systems such as water suppression.

- Backup suppression systems.

- Automatic UL rated fire dampers in HVAC system.

- Alternate shutdown capability.

I) Water Chemistry - Mr. Eric Yochheim, Chemistry Engineer, PSE&G, summarized the goals of the water chemistry control program as to protect the materials of construction from unwanted corrosion and to minimize radiation buildup; and as such to maximize availability and operating life of the plant. The control program is based on two main factors: Operation of the condensate polishing demineralizer and reactor water cleanup filter demineralizer in an optimum manner, and establishment of a strict chemistry control program that includes predetermined action level response program to off normal conditions.

J) Flood Generated Floating Missiles - Mr. Robert P. Douglas, Manager Licensing and Analysis, PSE&G, stated that, based on the analysis performed, PSE&G concludes that, the imapct of "floating marine missiles" need not be considered in the design bases for Hope Creek. The results of the analyses also indicate that all of the door structures on safety related structures are able to withstand the impact of the postulated non marine "floating missiles."

K) Environmental qualification of Equipment - Mr. Joseph Wroblewski, Principal Engineer, PSE&G, stated that PSE&G will qualify electrical equipment to comply with 10 CFR 50.49 and NUREG-0588. Equipment in a mild environment will be qualified. The program will be completed by September 1985. Maintenance and surveillance will ensure continued qualification. For harsh environment, the program consists of the following:

- Identify Equipment
- Determine environments caused by DBE
- Identify equipment to be qualified
- Specify qualification requirements
- Review and approve vendor qualification

- Compile NRC audit packages
- Compile maintenance and surveillance requirements

Mr. Ebersole expressed his concerns regarding (a) the harsh environments (hostile conditions) inside containment, and (b) the hostile conditions that occur in normal environment when there is steam line failures where there is prolonged discharge and the atmosphere is saturated steam.

The applicant replied that the equipment of interest inside containment would be terminal blocks associated with motor-operated valves for containment isolation and these are currently not vented. The applicant's program would actually simulate such conditions, even for a vented dip hole conditions, this would be accounted for in the qualification program and is monitored during testing.

Mr. Michelson asked if the applicant is in agreement with the statement in the SER which states that all Class 1E equipment has been either protected from or qualified for the environment that would be caused by inadvertent actuation of permanently installed fire suppression system. The applicant replied by saying yes; the requirement is that, if a threat exist in a certain area of the plant, the equipment will be protected.

L) AC Power Systems Reliability and Station Blackout

Mr. Pavincich, PSE&G, stated that the AC power for Hope Creek plant is very reliable and the loss of all AC power is a very low probability event. Mr. Michelson, responded by asking what action the applicant would take if it did happen. Mr. Pavincich replied by stating that the thought is to depressurize in a controlled manner to limit drywell temperatures, to reestablish reactor vessel inventory as needed with steam-driven HPCI and RCIC turbines to assure that the subsequent temperature rise in the necessary rooms requiring functional equipment is not excessive. Also, there is D.C. power to hold the unit in a stable condition so there is very positive assurance of restoration of A.C. Power. The applicant mentioned that, upon demand of four diesel generators, efforts are naturally directed towards the restoration of at least two of these units. One method toward restoration is a 35 Megawatt gas turbine at the Salem Station. The applicant has procedures to reestablish decay heat removal capability for both Salem units as well as the Hope Creek unit. There are station batteries designed to operate in excess of four hours. The applicant indicated that there is a very strong likelihood that the AC power will be restored within one hour.

M) Generic Material Problems - Mr. Joseph Rogozenski, Principal Engineer, PSE&G, provided an overview of four topics, namely: intergranular stress corrosion cracking, bolting materials in the ongoing surveillance program, fracture toughness of ferritic materials, and component and piping supports. Guidelines of NUREG-0313, Rev. 1., were incorporated where stainless steel was used. Further, Reg. Guides 1.31 & 1.44 were the bases for controlling the welding of stainless steel materials. All the counter actions taken on the piping and nozzles provides a positive deterrent to cracking at Hope Creek.

N) Operations, Staffing and Training - Mr. H. Denis Hanson, Manager of Nuclear training PSE&G provided an overview of the operation staffing layout and structure. He also described PSE&G resource commitment to training which includes:

- Technical and instructional capability qualification requirements for instructional staff
- line authorization for training and training support functions
- A full scope, Hope Creek simulator
- Extensive laboratories, shops, mock-ups, and audio/visual media
- Training center that was opened in August 1982.

The applicant described the training to mitigate core damage, which includes:

- Licensed Operator: To (a) recognize and mitigate effects of severe accidents, (b) meet requirements of NUREG 0737, and (c) classroom and lab/simulator training.

- Training non-licensed operations personnel to gain sufficient knowledge of Hope Creek systems, operations, and procedures.

The applicant also described the intraplant communications system for emergency operating conditions, which includes:

°Plant telephone network

°Voice paging

°UHF radio system

°Sound-powered system dedicated to alternate shutdown stations

III) NRC - Region I Presentation:

Richard Starostecki, Division Director of Project and Resident Programs, presented NRC Region I's evaluation of construction quality at Hope Creek. He stated that NRC Region I began performing inspections at Hope Creek in 1973, and has completed 140 inspections since that time. These inspections involved observations of work in progress, examination of completed work, independent measurements and calculations, and the examination of quality records. Mr. Starostecki described the inspection program, which includes:

- Inspection History
- Enforcement Record
- Regional Construction Team Inspection
- Independent Non-Destructive Examination
- Review of Construction Deficiencies
- Follow-up on Allegations
- SALP Reviews

Overall, Region I finds the Construction Program at Hope Creek to be acceptable. In addition, Region I review adds confidence that PSE&G, Bechtel, and the various subcontractors are capable of building a quality nuclear plant. The applicant has remained on schedule and close to budget.

IV) <u>Conclusions:</u>

The Subcommittee seemed to be pleased by the applicant's and the NRC Staff's presentations. Dr. Siess described the agenda for the full Committee meeting on Hope Creek to be held on December 13, 1984. Dr. Siess stated that he would like the applicant to give short presentations at the full Committee meeting for the following items:

- External hazards, limited to whatever have been of interest historically in connection with Hope Creek, including flood level and flood protection.

- Update on the river traffic situation, which was a question at the Construction Permit stage.

- Study regarding reduction in plant trips which includes feedwater related trips in BWRs and the efforts to reduce challenges to safety system including hardware and human factors.

- Radiation protection program.

- Seismic Design - Summary of liquefaction issue and any new information.

Mr. Michelson stated that he would like to have more information regarding the following items:

I. Isolation Valve Reliability (from LOCA outside primary containment for example, HPCI or RCIC steamline or reactor water cleanup suction line breaks). Require portions of the following document and information which are applicable to judging valve operability under external LOCA isolation conditions.

   °Valve type and size.

   °Purchase specifications and requirements (LOCA fluid flow, and phase conditions).

°Factory or laboratory tests or analysis requirements or results (tests, or analyses under blowdown conditions and how related to requirements).

°Valve adjustment (vendor, or utility instructions for torque switch, limit switch and stem packing adjustments--relationship to LOCA conditions ).

°Preoperational tests (how related to factory or laboratory tests and analyses).

°Technical specifications requirements (how related to factory or laboratory tests and analyses, or to measurements or torque margin available).

°Valve operating environment with or without LOCA isolation.

°Environmental qualifications.

°Subcompartment pressure buildup and consequences with and without LOCA isolation or with delayed or partial isolation (compared to structural capability).

°Accountability for aging effects.

°Protection of other ECCS systems from each postulated LOCA outside primary containment.

°How are valve control circuits wired relative to bypass of thermal overload at MCC for ECCS injection signal or other conditions (such as, LOCA outside primary containment).

II. Cask Drop Accidents. For each credible scenario:

°Probability of occurrence.

°Effect on plants and torus and attached piping (for example, suction to CS, RHR, HPCI, and RCIC). And possible loss of torus water.

°Water level in torus tunnel for case of hole in bottom of torus. (Due to tearout of suction lines). With loss of vapor suppression and consequential containment pressurization.

°Effect of hydrostatic pressure on ECCS pump compartments.

°Effects of torus flotation on structure (if credible).

Handling Other Heavy Loads.

Effects of worst case dryer or separator drops into vessel or storage pool.

Effect of worst case shield plug drop into vessel or storage pool (also considered effect on vessel or refueling collar).

III. Use of Fiberglass Insulation

°Type, particular size, and density of insulation.

°Extent of insulation disruption by LOCA inside containment.

°Effect of insulation on primary containment sump.

°Effect of insulation on RHR and CS pump bearing cyclone separator and pump bearings (other pumps?).

IV. Fire Protection in Diesel Generator Area

°Fire qualifications of offsite power bus duct.

°Possible scenarios for diesel generator runaway and ability to confine the incident.

V. Loss of All AC Power

HPCI and RCIC room heatup rate data and calculation.

Mr. Ebersole added that he would also like to have more information regarding:

° Environmental qualification of electrical equipment and what is the applicant prepared to do about moisture intrusion in the safety-related electrical equipment.

° Position on simplified "last resort" cooling system as used at Limerick or advanced BWR design.

- Fire protection, responses to outstanding items, and the potential for $Co_2$ overpressurization.

Dr. Siess commented that additional information is still needed from the NRC Staff regarding the following items:

- NRC Staff position on New Brunswick and Charleston earthquakes (seismic design)

- Summary of liquefaction issue

- Cask drop accidents, credible scenarios.

Adjournment.

NOTE: Additional meeting details can be obtained from a transcript of this meeting available published in the NRC Public Document Room, 1717-H Street, N.W., Washington, D.C., 20555, or can be purchased from ACE Federal Reporters, Inc., 444 North Capitol Street, Washington, D.C., 20001, (202) 347-3700.

ATTACHMENT A

## NUCLEAR REGULATORY COMMISSION

### Advisory Committee on Reactor Safeguards; Subcommittee on Hope Creek Generating Station Unit 1; Meeting

The ACRS Subcommittee on Hope Creek Generating Station Unit 1 will hold a meeting on November 28 and 29, 1984, at the Hilton of Philadelphia, Civic Center Blvd. and 34th Street, Philadelphia, PA

The entire meeting will be open to public attendance.

The agenda for subject meeting shall be as follows:

Wednesday, November 28, 1984–2:00 p.m. until the conclusion of business
Thursday, November 29, 1984–8:30 a.m. until the conclusion of business

The Subcommittee will review the operating license application of the Public Service Electric and Gas Company for the Hope Creek Generating Station.

Oral statements may be presented by members of the public with concurrence of the Subcommittee Chairman; written statements will be accepted and made available to the Committee. Recordings will be permitted only during those portions of the meeting when a transcript is being kept, and questions may be asked only by members of the Subcommittee, its consultants, and Staff. Persons desiring to make oral statements should notify the ACRS staff member named below as far in advance as practicable so that appropriate arrangements can be made.

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

The Subcommittee will then hear presentations by and hold discussions with representatives of the Public Service Electric and Gas Company, NRC Staff, their respective consultants, and other interested persons regarding this review. Further information regarding topics to be discussed, whether the meeting has been cancelled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefore can be obtained by a prepaid telephone call to the cognizant ACRS staff member, Dr. Medhat M. El-Zeftawy (telephone 202/634-3267) between 8:15 a.m. and 5:00 p.m., est. Persons planning to attend this meeting are urged to contact the above named individual one or two days before the scheduled meeting to be advised of any changes in schedule, etc., which may have occurred.

Dated: November 5, 1984.

Morton W. Libarkin,

*Assistant Executive Director for Project Review.*

[FR Doc. 84-29844 Filed 11-8-84; 8:45 am]
BILLING CODE 7590-01-M

**Advisory Committee on Reactor Safeguards Subcommittee on Hope Creek Generating Station Unit 1; Location Change**

The ACRS Subcommittee on Hope Creek Generating Station Unit 1 scheduled at the Hilton of Philadelphia has been changed to the *Sheraton University City, 36th & Chestnut Street, Philadelphia, PA for November 28 and 29, 1984.* Notice of this meeting was published Friday, November 9, 1984 (49 FR 44832).

The entire meeting will be open to public attendance.

The agenda for subject meeting shall be as follows:

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Dated: November 15, 1984.

Thomas G. McCreless,

*Assistant Executive Director for Technical Activities.*

[FR Doc. 84-30551 Filed 11-20-84; 8:45 am]

BILLING CODE 7590-01-M

ATTACHMENT B

- PRESENTATION SCHEDULE -

ACRS SUBCOMMITTEE MEETING
HOPE CREEK GENERATING STATION UNIT 1

NOVEMBER 28 AND 29, 1984
PHILADELPHIA, PENNSYLVANIA

WEDNESDAY - NOVEMBER 28, 1984

| | SUBJECT | PRESENTER | BOOK SECTION | ACTUAL TIME |
|---|---|---|---|---|
| 1. | OPEN REMARKS | C. Siess | | 2:00 - 2:10 pm |
| 2. | NRC STAFF PRESENTATION | | | |
| | a. Major differing Technical Issues and the Schedule for their resolution | D. Wagner | | 2:10 - 3:10 pm |
| | b. Construction Experience-Noncompliance during construction | | | |
| | c. Significant SER Open Items, confirmatory issues and licensing conditions | | | |
| 3. | APPLICANT'S PRESENTATION | | | |
| 3.1 | a. Introduction | R. Eckert | | 3:10 - 3:15 pm |
| | b. Overview of plant layout and principal design features - proximity to Salem | W. Gailey | 1 | 3:15 - 3:20 pm |
| | c. Construction Status and Plant Start-up Schedule | P. Landrieu | 2 | 3:20 - 3:25 pm |
| | d. Plans to handle maintenance, inservice inspection and pre-operational testing, including recordkeeping and document control | S. LaBruna | 3 | 3:25 - 3:30 pm |
| | e. Resolution of comments from previous ACRS letters | W. Gailey | 4 | 3:30 - 3:35 pm |

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| SUBJECT | PRESENTER | BOOK SECTION | TIME |
|---|---|---|---|
| 3.2 Organizations and Management | | | |
| a. Corpor·· Organization | R. Eckert/<br>T. Martin | 5<br>6 | 3:35 - 4:35 pm |
| b. Nuclear Organization | R. Uderitz | 7 | |
| c. Safety Review Committees, compliance with NUREG-0731, "Management Structure and Technical Resources" | R. Uderitz | 8 | |
| d. Current Status of Staffing (Engineering, Management and other key personnel) | R. Uderitz | 9 | |
| e. Nuclear-Related Operating Experience of key personnel | R. Salvesen | 10 | |
| f. Feedback of operating experience to operators and other key operations staff | S. La Bruna | 11 | |
| ***** BREAK ***** | | | 4:35 - 4:45 pm |
| 3.3 Seismic Design of Plant and Equipment | C. Churchman | 12 | 4:45 - 5:15 pm |
| Comments on seismic margins-site specific spectra, liquefaction | | | |
| 3.4 Control Room and I&C Systems | | | |
| a. Control Room Design, Post-Accident Habitability and remote shutdown capability | Y. Yaworsky | 13 | 5:15 - 5:30 pm |
| b. Human Factors Review | T. Mc Guire | 14 | 5:30 - 5:45 pm |
| c. Post-Accident monitoring-conformance with Reg. Guide 1.97, Revision 2 | | 15 | 5:45 - 6:00 pm |

| SUBJECT | PRESENTER | BOOK SECTION | TIME |
|---|---|---|---|
| **HURSDAY, NOVEMBER 29, 1984** | | | |
| .5 Introduction | C. Siess | | 8:30 - 8:35 am |
| .6 Fire Protection | W. Pavincich | 16 | 8:35 - 9:00 am |
| .7 Water Chemistry | J. Nichols | 17 | 9:00 - 9:15 am |
| .8 Environmental Qualification of Equipment | J. Wroblewski | 18 | 9:15 - 9:30 am |
| .9 ATWS Mitigation | T. Mc Guire | 19 | 9:30 - 9:45 am |
| a. Compliance with the proposed ATWS rule | | | |
| b. ATWS independent initiation | | | |
| c. Response to Salem incident-views on modification or testing of SCRAM breaker configuration | | | |
| 3.10 AC Power Systems Reliability and Station Blackout | W. Pavincich | 20 | 9:45 - 10:15 am |
| ***** BREAK ***** | | | 10:15 - 10:25 am |
| 3.11 Response to Generic Material Problems | J. Rogozenski | 21 | 10:25 - 10:55 am |
| 3.12 Flood Generated Floating Missiles | R. Douglas | 22 | 10:55 - 11:10 am |
| 3.13 Quality Assurance | | | |
| a. Overview of Policy and Organization | R. Eckert | 23 | 11:10 - 11:15 am |
| b. Quality Control problems experienced during construction and their resolution | A. Giardino | 24 | 11:15 - 11:45 am |
| c. Operational Quality Assurance | C. Johnson | 25 | 11:45 - 12:00 noon |
| ***** LUNCH ***** | | | 12:00 - 1:00 pm |

| SUBJECT | PRESENTER | BOOK SECTION | TIME |
|---|---|---|---|
| 3.14 Emergency Planning | D. Moeller | 26 | 1:00 - 1:15 pm |
| 3.15 Fitness for Duty and Personnel Selection | S. Kosierowski | 27 | 1:15 - 1:30 pm |
| 3.16 Operations Staffing and Training | | | 1:30 - 2:30 pm |
| a. Training of operators, auxiliary operators and maintenance personnel | H. Hanson | 28 | |
| b. Training to handle severe accidents - emergency operating procedures | H. Hanson/ S. LaBruna | 29 | |
| c. Communications during normal as well as emergency situations | W. Pavincich | 30 | |
| 3.17 Radiation Protection Program | W.Britz/ R. Lovell | 31 | 2:30 - 2:50 pm |
| 3.18 Summary | R. Eckert | | 2:50 - 3:05 pm |
| 4. CAUCUS | C. Siess | | 3:05 - 3:30 pm |

***** ADJOURNMENT *****

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ATTACHMENT C

Confirmatory Issues

| Issue | | SER section |
|---|---|---|
| (1) | Feedwater isolation check valve analysis | 3.6.2 |
| (2) | Plant-unique analysis report | 3.9.3.1, 6.2.1.7 |
| (3) | Inservice testing of pumps and valves | 3.9.6 |
| (4) | Fuel assembly accelerations | 4.2 |
| (5) | Fuel assembly liftoff | 4.2 |
| (6) | Review of stress report | 5.2.1.1 |
| (7) | Use of Code cases | 5.2.1.2 |
| (8) | Reactor vessel studs and fasteners | 5.3.1.5 |
| (9) | Containment depressurization analysis | 6.2.1.4 |
| (10) | Reactor pressure vessel shield annulus analysis | 6.2.1.5.1 |
| (11) | Drywell head region pressure response analysis | 6.2.1.5.2 |
| (12) | Drywell-to-wetwell vacuum breaker loads | 6.2.1.7 |
| (13) | Short-term feedwater system analysis | 6.2.3 |
| (14) | Loss-of-coolant-accident analysis | 6.3.5, 15.9.3 |
| (15) | Balance-of-plant testability analysis | 7.2.2.3 |
| (16) | Instrumentation setpoints | 7.2.2.5 |
| (17) | Isolation devices | 7.2.2.6 |
| (18) | Regulatory Guide 1.75 | 7.2.2.7 |
| (19) | Reactor mode switch | 7.2.2.9 |
| (20) | Engineered safety features reset controls | 7.3.2.6 |
| (21) | High pressure coolant injection initiation | 7.3.2.9 |
| (22) | IE Bulletin 79-27 | 7.4.2.1 |
| (23) | Bypassed and inoperable status indication | 7.5.2.4 |
| (24) | Logic for high pressure coolant injection interlock circuitry | 7.6.2.1 |
| (25) | End-of-cycle recirculation pump trip | 7.6.2.4 |

(Continued)

| Issue | SER section |
| --- | --- |
| (26) Multiple control system failures | 7.7.2.1 |
| (27) Relief function of safety/relief valves | 7.7.2.2 |
| (28) Main steam tunnel flooding analysis | 8.3.3.1.4 |
| (29) Cable tray separation testing | 8.3.3.3.2 |
| (30) Use of inverter as isolation device | 8.3.3.3.4 |
| (31) Core damage estimate procedure | 9.3.2 |
| (32) Continuous airborne particulate monitors | 12.3.4.2 |
| (33) Qualifications of senior radiation protection engineer | 12.5.1 |
| (34) Onsite instrument information | 12.5.2 |
| (35) Airborne iodine concentration instruments | 12.5.2 |
| (36) Emergency Plan items | 13.3.2.1, 13.3.2.4-9, 13.3.2.12, 13.3.2.15 |
| (37) TMI Item II.K.3.18 | 15.9.3 |

ATTACHMENT D

License Conditions

| License condition | | SER section |
|---|---|---|
| (1) | Turbine system maintenance program | 3.5.1.3.3 |
| (2) | NUREG-0803 implementation | 4.6 |
| (3) | Inservice inspection | 6.6 |
| (4) | Postaccident sampling system | 9.3.2 |
| (5) | Solid waste process control program | 11.4.2 |
| (6) | Partial feedwater heating | 15.1 |
| (7) | Cask drop accident | 15.7.5 |