Date issued:
ACRS-1768
OCT 10 1980

MINUTES OF THE SEQUOYAH SUBCOMMITTEE
MEETING
JULY 9, 1980
WASHINGTON, DC

The ACRS Subcommittee on the Sequoyah Nuclear Plant, Units 1 and 2 met on July 9, 1980, to discuss the Tennessee Valley Authority (TVA) application for a license to operate the Sequoyah Nuclear Plant, Units 1 and 2 at full power. Presentations were given by the NRC Staff, TVA, and the Westinghouse Electric Corporation. The principal topics which were discussed were the status of the NRC Staff's open review items, the implementation of the NTOL requirements, hydrogen control, and the Applicant's and NRC Staff's work on risk assessment for ice condensers and the applicability of filtered vented containment. The Subcommittee also discussed the differing professional opinion of the adequacy of the review of weld repairs made to the pressurizer relief pipe on Unit 1. Notice of this meeting was published in the Federal Register on June 24, 1980. A copy of this notice is included as Attachment A and a list of attendees is included as Attachment C. Portions of the material provided to the Subcommittee at this meeting are included as Attachment D. The complete set of material provided to the Subcommittee is in the ACRS files. No oral statements were given by members of the public nor were there any requests for time to make oral statements. No written statements were submitted. A transcript was kept of the Subcommittee meeting proceedings.

The ACRS members in attendance were Dr. J. Carson Mark, Subcommittee Chairman, and Mr. W. Mathis. The ACRS consultants present were Dr. I. Catton, Dr. W. Lipinski, and Dr. Z. Zudans, and Dr. R. Savio of the ACRS Staff was also present. Dr. Savio was the Designated Federal Employee for this meeting. The entire meeting was held in open session.

STATUS OF THE NRC REVIEW - C. Stahle, NRC Staff

Mr. Stahle summarized the status of the open review issues currently under consideration. The Staff has listed 13 full-power non-TMI issues and 40 full-power TMI related issues for Sequoyah. Actions on 8 of the non-TMI issues and on 15 of the TMI related items are essentially complete. Thirteen of the TMI related items are to be resolved by agreed upon dates and one of these items (degraded core) will be resolved in rulemaking. Actions on all remaining incomplete items are expected to be completed within the next two to three weeks. Listings of these items are on pages 1-3 of Attachment D.

Mr. Stahle noted that there was the difference of professional opinion on the part Mr. John Halapatz of the NRC Staff regarding the adequacy of the Sequoyah Unit 1 weld repair of the pressurized relief pipe. The pressurizer relief pipe had been bent during the hot functional test as a result of a failure of the relief pipe to slide through a pipe hanger support. The pipe in question is fabricated from six-inch 316SS pipe. The pipe was repaired by grinding two 270 degree, two-thirds thickness depth grooves in the pipe opposite to and straddling the bent section. The grooves were filled with weld material, reground to remove the weld material and then filled a second time with weld material. Weld metal shrinkage provided the forces to straighten the affected section of the pipe. The pipe was not subjected to a hydrostatic test after the weld repair. ASME codes would have required that the test be performed had the pipe section been penetrated. Mr. Halapatz's concerns were related to the lack of hydrostatic testing and the possibility that the pipe material in the area of the weld may have been sensitized. The matter was discussed at some length. It was generally agreed that, in view of the degree of inspection the weld area had received, a hydrostatic test of the system would not increase the assurance that the system would perform as designed. The issue of pipe metal sensitization was not, however, resolved. Construction of and examination of a prototypical mock-up and third party inspection were suggested as ways of resolving the dispute as to the adequacy of the pipe system after the repair. Mr. Halapatz has summarized his concerns in the document included as Attachment F.

FLOOD PROTECTION - M. Burzynski, TVA

Mr. Burzynski indicated that the probable maximum flood for the Sequoyah plant was based on a three day storm, occurring over the 21,400 sq. mile of watershed, with a total rain fall of 16.8 inches and preceded by a three day storm occurring three days earlier with a total rainfall of 6.7 inches. The seismic design base flood was based on the failure of four upstream dams coincident with a flood crest equal to that of one-half the probable maximum flood. The flood protection plan calls for the immediate controlled shutdown and cooldown, the call-up of additional personnel, and the activation of the diesel generator system. These steps would be taken when the flood warning is received. Following this the flood protection plan calls for using the high pressure fire protection system as a replacement for the auxiliary feedwater system, replacing

component cooling water with the essential raw coolant water, filling radwaste tanks to prevent flotation, sealing drains to the diesel generator and emergency raw coolant water buildings, and disconnecting all batteries situated below the design base flood level. The flood warning system provides a minimum of 24 hours notice before the flood crest reaches the plant. This would allow implementation of all parts of the flood protection plan.

SINGLE UNIT/TWO UNIT DESIGN CHANGES - D. Williams, TVA

Mr. Williams summarized the design changes which would be implemented in the Sequoyah plant before two unit operation. TVA was required to expand their service water heat dissipation facility at Sequoyah as a result of the passage of the Federal Water Pollution Control Act of 1972. New natural draft cooling towers were added to allow full power operation under warm weather conditions. The design of these cooling towers is such that they discharge into the condenser water intake pumping station. Design studies were performed which show that under certain conditions, the discharge from the natural cooling towers would exceed the design temperature of the ERCW pump. A decision was made to build a new ERCW pumping station. The present pumping station is adequate to accommodate the operation of Unit 1. The new pumping station will be for the operation of Unit 2. In addition, an interim auxiliary building secondary containment enclosure has been added to provide a more effective barrier to airborne contaminants. This barrier will no longer be needed after the completion of the construction associated with Unit 2.

STATUS OF THE LOW POWER TEST PROGRAM - C. Stahle, NRC

Mr. Stahle indicated that the review of the low power test program was essentially completed and that approval of the program was expected by July 11, 1980. A similar program has been approved for North Anna Unit 2 and is currently underway.

REACTOR VESSEL NOZZLE UNDER CLAD CRACKING - E. Toddy, Westinghouse

Mr. Toddy indicated that the concerns relative to the existence of cracking in the Sequoyah reactor vessel nozzles were related to the discovery by the Westinghouse-French licensee of the cracking in pressure vessels manufactured in Europe. The cracking is believed to be hydrogen-induced and as a result of the welding process/heat treatment used in applying the cladding. The

Sequoyah Unit 1 nozzle was manufactured in Rotterdam and was accordingly inspected. Underclad cracking was found in the vessel nozzles. All cracks were below the designated ASME critical flaw size but were quite numerous in some sections of the nozzles. The Prairie Island Unit 1 vessel nozzles were clad using similar techniques. This vessel will be inspected in about six months at the refueling outage. The vessel has been in operation for about nine years. Information on the existence of underclad cracking and propagation should be obtained from this inspection.

STATUS REPORT ON ICE CONDENSER RISK ASSESSMENT STUDIES - R. Christe, TVA

Mr. Christe indicated that there were four programs currently underway. The Systems Interaction Methodology Applications Program being conducted at Sandia under NRC sponsorship is using the Watts Bar plant for the study and is similar to the Sequoyah plant. It was concluded that the facility was well protected against interactions which were considered within the scope of the study.

The Reactor Safety Study Methodology Applications Program is being conducted by RES-PAS. The objective is to determine dominant accident sequences using the methodology developed in WASH-1400. The plants being studied are Sequoyah, Grand Gulf, Calvert Cliffs, and Oconee. The Sequoyah ice condenser study is not yet completed. The results at this point indicate that ice condenser plants have at different dominant accident sequences but the risk associated with the plant is similar to what would be expected in larger dry containment plants.

TVA has contracted with Kaman Sciences, Inc. for the performance of the reliability evaluation of the Sequoyah Unit 1 auxiliary feedwater system. The GO code was employed in this analysis. The GO code had been used extensively for defense applications studies and is oriented to success tree rather than fault tree sequences. The results indicated that the probability of successfully starting an auxiliary feedwater system upon demand and providing adequate water fluid pressure to at least two steam generators was 0.99999. In a loss-of-offsite power with diesel generators and battery backup available, the auxiliary feedwater system supply success probability was calculated at 0.99997.

In addition, TVA intends to use the GO methodology to assess plant availability and plant safety. Phase 1 of this work is to be conducted with simplified plant models. It was initiated on July 1, 1980 and is expected to be completed by December 31, 1980. Phase 2 which involved expansions of the simplified model and the application of improved reliability data will be initiated on January 1, 1981 and will be completed within a year.

STATUS OF PAS WORK ON ICE CONDENSER RISK ASSESSMENT - M. Taylor, NRC/PAS

Mr. Taylor described the ice condenser reliability study being conducted within the Reactor Safety Study Methodology Applications Program. This program involved a reassessment of the WASH-1400 work to obtain an improved baseline for the WASH-1400 model plants and studies more current LWR designs, one of which was the Sequoyah ice condenser. The estimated probability of severe core damage resulting in these studies is given on Table 1 in Attachment D. It is noted that the risk associated with the Sequoyah plant is comparable to that associated with the Surry and Peach Bottom WASH-1400 plants. Site characteristics are incorporated in these estimates. The Indian Point studies involved risk estimates for the Indian Point site with various reactor types. A summary of these results is given on pages 4-5 of Attachment D. Mr. Taylor noted that the ice condenser failure most affecting risk is the overpressure failure and that, while the ice condenser dominant risk sequences are somewhat different from the WASH-1400 PWR and PWR designs, the overall risk seems to be comparable within the uncertainties of the analysis.

STATUS ON HYDROGEN CONTROL STUDIES - G. Dilworth, TVA and W. Butler, NRC

Mr. Dilworth summarized the work that TVA had done on hydrogen control in ice condensers. Mr. Dilworth noted that the studies had been conducted over the past nine month period and had concluded that the current Sequoyah design can withstand substantial amounts of hydrogen above the design basis. The maximum containment strengths of 33 psig and 42.5 psig, based respectively on the yield and ultimate strengths, have been caluclated. TVA has concluded that the present Sequoyah design can withstand the hydrogen generated at 25% metal-water reaction of the ultimate strength is used as the failure criteria. The assumptions are that the hydrogen is generated and released into the containment and that a fast burn (5 to 30 seconds) occurs and that no shock wave is associated with the hydrogen burn. This would be similar to the postulated

TMI-2 scenario. A number of concepts for providing additional protection against hydrogen burns were studied. Vented containments using filtered release, additional containment shells, and coupled containment structures (Unit 1 and Unit 2) were studied. Controlled combustion (ignition sources) and means for preventing combustion (nitrogen inerting and halon suppression) were also studied.

Filtered vented containments were judged not to be effective for rapid pressure transients and to have associated with them a risk of unnecessarily releasing contamination. The coupled containment concept studied was that of connecting the containments for Unit 1 and Unit 2 through a valved tunnel. These were judged not to be effective for rapid pressure transients and to have the potential for degrading the safety of the second unit in the event of an accident. Controlled combustion sources were judged to have the highest potential for reducing the risk from hydrogen during most accidents leading to clad oxidation. The system also has the advantage of having a moderate initial cost and low operating and maintenance costs. Nitrogen inerting was judged to be impractical for an ice condenser containment. The containments are small and require frequent entry for inspection and maintenance duties. Backfits which would eliminate the need for frequent entry would be difficult and expensive. Halon suppression was judged to be potentially effective in preventing hydrogen combustion. The decomposition products, however, may degrade the long term operability of the plant equipment under accident conditions. The most promising concepts for enhancing the design capability for hydrogen control were ignition sources and halon suppression.

TVA proposed that a distributed ignition source system be installed as an interim hydrogen control measure. Programs would be continued which would be directed at improving the distributed ignition source system and to further the applicability of halon suppression. Mr. Dilworth indicated that the distributed ignition source system had the potential for controlling hydrogen up to 70% metal-water reaction. It is expected that the core could not be contained within the pressure vessel beyond this point.

Mr. Butler indicated that the Staff would review the proposed ignition system. TVA would be permitted to proceed with the installation of the system but not to operate it until NRC concurrence was obtained. It is expected that the

installation of this system and the NRC review would take about three months. In addition, Mr. Butler indicated that the NRR had initiated a user's request for a safety research program directed to evaluating systems for mitigation of degraded core/core melt accidents. This program would be directed toward the development of information on mitigation systems for all LWR containments for use in the upcoming rulemaking proceeding. The short term program (6-12 months) would be directed toward evaluating systems for the mitigation of degraded core accidents in ice condensers and Mark III containments. Hydrogen generation rates for degraded core accidents would be evaluated and containment response during a hydrogen burn would be determined. The effectiveness of various hydrogen control systems under these conditions would be studied. The long term program (2 years) would be directed towards the study of venting systems and advanced hydrogen control systems for all LWR containments. Distributed ignition sources, large thermal recombiners, halon systems, inerting systems, water fog systems, large catalytic recombiners, and oxygen scavenging systems would be studied.

STATUS REPORT ON FILTERED/VENTED CONTAINMENT SYSTEMS - J. Meyer, NRC/NRR

Mr. Meyer described the Staff's current studies on filtered vented containemnt systems. These studies were directed toward the Zion and Indian Point plants. Conceptual designs for filtered vented containments have been studied. These involve venting to water suppression pools with the possible addition of follow-on sand and gravel and charcoal/HEPA filters. Systems would not be able to cope with all ranges of hydrogen release unless very large vent lines were used. Vent lines of about a three foot diameter would have the capability of dealing with most accident sequences.

****

NOTE: For additional details, a complete transcript of the meeting is available in the NRC Public Document Room, 1717 H Street, N.W., Washington, D.C. 20555 or from the Alderson Reporting Co., Inc., 300 7th Street, S.W., Reporters Building, Washington, D.C. 20024, (202/554-2345).

ed during the balance of the

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

Further information regarding topics to be discussed, whether the meeting has been cancelled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by a prepaid telephone call to the cognizant Designated Federal Employee, Mr. Peter Tam (telephone 202/634-1413) between 8:15 a.m. and 5:00 p.m., EDT.

Dated: June 19, 1980.

John C. Hoyle,

Advisory Committee, Management Officer.

[FR Doc. 80-18984 Filed 6-23-80; 8:45 am]

BILLING CODE 7590-01-M

Advisory Committee on Reactor Safeguards, Subcommittee on the Sequoyah Nuclear Plant; Meeting

The ACRS Subcommittee on the Sequoyah Nuclear Plant will hold a meeting on July 9, 1980, in Room 1046, H St., NW, Washington, DC to w the Tennessee Valley Authority A) request for a license to operate this plant. Notice of this meeting was published June 20, 1980.

In accordance with the procedures outlined in the Federal Register on October 1, 1979, (44 FR 56408), oral or written statements may be presented by members of the public, recordings will be permitted only during those portions of the meeting when a transcript is being kept, and questions may be asked only by members of the Subcommittee, its consultants, and Staff. Persons desiring to make oral statements should notify the Designated Federal Employee as far in advance as practicable so that appropriate arrangements can be made to allow the necessary time during the meeting for such statements.

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

The agenda for subject meeting shall e as follows:

Wednesday, July 9, 1980, 8:30 a.m. until the conclusion of business.

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

The Subcommittee will then hear presentations by and hold discussions with representatives of the NRC Staff, the Tennessee Valley Authority (TVA), their consultants, and other interested persons.

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

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

Dated: June 18, 1980.

John C. Hoyle,

Advisory Committee, Management Officer.

[FR Doc. 80-18985 Filed 6-23-80; 8:45 am]

BILLING CODE 7590-01-M

[Docket No. 50-32]

Metropolitan Edison Co. (Three Mile Island Nuclear Station, Unit 2); Issuance of Director's Decision Under 10 CFR 2.206

On September 14, 1979, a notice was published in the Federal Register that a petiton by the Anti-Nuclear Group Representing York (ANGRY) was being considered under 10 CFR 2.206. ANGRY's petition requested that the Commission prepare an environmental impact statement concerning the venting of radioactive gases from the reactor buildng of the Three Mile Island Nuclear Station, Unit 2. Because this action will not cause any significant environmental impact, it has been determined not to prepare an environmental impact statement. Accordingly, ANGRY's petition is denied.

A copy of the formal decision denying the petition is available for inspection in the Commission's Public Document Room at 1717 H Street NW., Washington, D.C. 20555 and in the local public document rooms at the State Library of Pennsylvania (Government Publications Section), Education Building, Commonwealth and Walnut Streets, Harrisburg, Pennsylvania 17126 and the York College of Pennsylvania, Country Club Road, York, Pennsylvania 17405. A copy will also be filed with the Secretary for the Commission's review in accordance with 10 CFR 2.206(c). As provided in 10 CFR 2.206(c) this decision will become the final action of the Commission twenty days after issuance unless the Commission elects to review the decision on its own motion within that time.

Dated at Bethesda, Maryland, this 13th day of June, 1980.

For the Nuclear Regulatory Commission.

Edson G. Case,

Acting Director, Office of Nuclear Reactor Regulation.

[FR Doc. 80-18986 Filed 6-23-80; 8:45 am]

BILLING CODE 7590-01-M

DEPARTMENT OF TRANSPORTATION

Federal Railroad Administration

Public Meeting To Outline and Discuss Proposed Guidelines and Procedures Regarding Rock Island Railroad and Employee Assistance Act and Milwaukee Railroad Restructuring Act

On Wednesday, June 25, 1980, at 10:00 a.m., the Federal Railroad Administration (FRA) will hold a meeting in Room 8334 of the Nassif Building, 400 7th Street, Southwest, Washington, D.C. to outline and discuss the proposed guidelines and procedures to be issued by the Department of Transportation (DOT), under which the public may submit applications for directed service under section 104 of the Rock Island Railroad Transition and Employee Assistance Act (Pub. L. 96-254) and section 18 of the Milwaukee Railroad Restructuring Act (45 U.S.C. 916), and the proposed criteria which the Department will use to evaluate those applications.

The meeting is open to the public, including interested states and organizations who are considering applying for directed service under Pub. L. 96-254.

Issued in Washington, D.C., on June 20, 1980.

Michael T. Haley

Acting Chief Counsel.

[FR Doc. 80-19115 Filed 6-20-80; 2:08 pm]

BILLING CODE 4910-06-M

DRAFT AGENDA
ARSR PRESENTATION
TO
ACRS
WG-6 ON JULY 9, 1980

8:30 - 8:40	EXECUTIVE SESSION
8:40 - 9:00	INTRODUCTION - C. KELBER, NRC
9:00 - 9:30	REACTOR SAFETY MODELING AND ASSESSMENT - HUMMEL, ANL
9:30 - 10:00	3-D CODE DEVELOPMENT - SHA, ANL
10:00 - 10:45	SSC CODE DEVELOPMENT AND TESTING - GUPPY, BNL
	BREAK (10 MINUTES)
10:55 - 11:15	THERMALHYDRAULIC LMFBR SAFETY EXPERIMENTS - GINSBURG, BNL
11:15 - 11:40	AEROSOL MEASUREMENTS AND MODELING FOR FAST REACTOR SAFETY - GIESEKE, BCL
11:40 - 12:30	AEROSOL RELEASE AND TRANSPORT FROM LMFBR FUEL - KRESS, ORNL

MEETING DATE: ______ 980

SUBCOMMITTEE MEET... OYAH NUCLEAR PLANT

LOCATION: ROOM

J. C. Mark
W. Mathis
I. Catton
W. Lipinski
Z. Zudans
R. Savio
~~C. Stahle NRC~~

ATTENDANCE LIST

PLEASE PRINT

	NAME		AFFILIATION
1.	[illegible]		NRC - [illegible]
2.	[illegible]		[illegible]
3.	J. S. Wermiel		NRC - NRR - ASB
4.	Gary Zech	✓	NRC-NRR
5.	C. Stahle	✓	"
6.	[illegible]	✓	" "
7.	[illegible]	✓	" "
8.	T [illegible]		" "
9.	J [illegible]	✓	" "
10.	T KENYON		" "
11.	J Rajan		" "
12.	P. St[illegible]		" "
13.	RJ SERBU		" "
14.	O.D PARR	✓	" "
15.	M. M. Mendonca		" "
16.	R. H. Wessman	✓	NRC · IE
17.	J J Buzy	✓	" - NRR
18.	J R Beynum		TVA
19.	BOB LEYSE	✓	EPRI
20.	Mark Burzynski	✓	TVA
21.	ED MERRICK		TVA
22.	JERE BALLENTINE	✓	TVA
23.	L M. MILLS	✓	TVA
24.	[illegible] J. Esposito	✓	Westinghouse
	P. Gamble	✓	NRC

EETING DATE:

UBCOMMITTEE MEETING: SEQUOYAH NUCLEAR PLANT

OCATION: ROOM 1046

ATTENDANCE LIST

PLEASE PRINT

	NAME		AFFILIATION
1.	GEORGE DILWORTH	✓	TVA
2.	David Lambert	✓	TVA
3.	DON L WILLIAMS	✓	TVA
4.	[illegible]		TVA
5.	GARY R REED	✓	TVA
6.	Wang Lau	✓	TVA
7.	Ron Gamble		NRC
8.	MIKE SIAND		W
9.	[illegible]		[illegible] Hill
10.	GARY AUGUSTINE		W
11.	DAVE GOESER		W
12.	W. J. Johnson	✓	Westinghouse
13.	W R SPEZIALETTI		W
14.	J.M GRANT		NRC
15.	M.L BOYLE		NRC
16.	M.H. SCHWARTZ		PICKARD, LOWE & GARRICK (for ALAB)
17.	S.V. Asselin		Technology for Energy Corp
18.	C. L. STORLI		W
19.	T H Heitman		Duke Power Co
20.	W. H. Rasin		Duke Power Co
21.	T F Timmons		W
22.	D.C. AABYE		OPS
23.	Marc Silverstein		Wash Report + Press
24.	Marcia Edwards		Nuclear Tech., Inc.

EETING DATE: JULY 9, 198

UBCOMMITTEE MEETING: SEQUOYAH NUCLEAR PLANT

OCATION: ROOM 1046

ATTENDANCE LIST

PLEASE PRINT

NAME	AFFILIATION
T. L. Johnson	NRC
H. B. CLAYTON	NRC /NRR
).	
l.	
2.	
3.	
4.	
5.	
6.	
7.	
8.	
9.	
0.	
21.	
22.	
23.	
24.	

INCOMPLETE NON-TMI ISSUES ON SEQUOYAH UNIT NO. 1

1. Seismic Audit per ACRS Letter
2. Position Required Regarding Foudation Monitoring on Settlement
3. Position Required on Containment SUMP Debris
4. ECCS Evaluation Model Concerning Fuel Clad Swelling
5. Position Required Regarding Process Control Program
6. Equip. Qualifications Comply With the Guidelines of NUREG-0588
7. PAD 3-3 Performance Code - Complete Evaluation Regarding Restriction In the Use of this Code
8. ATWS - Review and Approve Operating Procedures
9. Compliance of IE Bulletin 79-27, Loss of Non-Class IE Instrumentation & Control Room System During Operation
10. Diesel Generator Reliability - Compliance with R.G. 1.108 and NUREG/CR-0660
11. Topical Reports WCAP-9226, 9230 and 9236 Related to Main Steam & Feedline Break Accidents
12. Q-List Complete Review of "Q-List" Requirements
13. Compliance of OIE Bulletin 80-06 Related to By-Pass. Override, Reset Circuits

1 2

INCOMPLETE (FULL-POWER) TMI ISSUES ON SEQUOYAH UNIT NO. 1

1. SHIFT TECH ADVISOR -1/81
2. IMMED. UPGRADE OF SRO & RO QUAL.-8/80
3. ADMIN. OF TRAINING PROGRAM FOR LICENSING EXAMS - 8/80
4. REV. SCOPE & CRITIERA FOR NORMAL LICENSING EXAMS - 8/80
5. REV. SCOPE & CRITERIA FOR SIMUL. EXAMS
6. PROC. FOR VEIRIFICATION OF CORRECT PERF. OF OP. ACTIVITIES
7. CONTROL ROOM DESIGN REVIEW
8. REACTOR COOLANT SYSTEMS VENTS- 1/81
9. POST-ACCIDENT SAMPLING - 1/81
10. TRAINING FOR MITIGATING CORE DAMAGE
11. ANALYSIS OF HYDROGEN CONTROL
12. DEGRADED CORE - RULEMAKING
13. RELIEF AND SAFETY VALVE TEST REQ.- 6/81
14. AFW RELIABILITY EVALUATION
15. AFW INITIATION AND INDICATION - 1/81
16. CONTAINMENT DEDICATED PENETRATION - 1/81
17. CONTAINMENT ISOLATION DEPENDABILITY
18. ADD. ACC. MONITORING INSTRUMENTATION - 1/81
19. INADEQUATE CORE COOLING INSTRUMENTS- 1/81
20. FINAL RECOM. OF B&O TASK FORCE
21. UPGRADE EMERGENCY PREPARDNESS
22. UPGRADE EMERGENCY SUPPORT FACILITIES- 1/81
24. COMMUNICATIONS
25. IMPL. OF NRC AND FEMA RESPON.
26. OFFSITE DOSE MEASUREMENTS
27. IN-PLANT RADIATION MONITORING- 1/81
28. CONTROL ROOM HABITABILITY
29. POWER - ASCENSION TEST

COMPLETE (FULL POWER) TMI ISSUES ON SEQUOYAH UNIT NO. 1

1. REACTOR INSPECTOR AT OPERATING REACTORS
2. SHORT TERM ACC. ANALYSIS AND PROC. REVISION
3. NSSS VENDOR REVIEW OF PROC.
4. PILOT MONITORING OF SELECTED EMERG. PROC. FOR NTOL APP.
5. LOW POWER TESTING TRAINING
6. PLANT SHIELDING
7. EMERG. POWER FOR PRESSURIZER HEATERS
8. PRIMARY COOLANT SOURCES OUTSIDE CONTAINMENT

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

(7)

June 16, 1980

Docket Nos. 50-327/328

MEMORANDUM FOR: S. S. Pawlicki Chief
Materials Engineering Branch
Division of Engineering

FROM: J. Halapatz
Materials Engineering Branch

SUBJECT: EXPRESSION OF DIFFERING PROFESSIONAL OPINION IN THE MATTER OF THE ADEQUACY OF SEQUOYAH UNIT ONE WELD DRAWBEAD REPAIR OF PRESSURIZER RELIEF PIPE

The author of this memorandum, hereinafter referred to as the minority, herewith expresses his minority opinion in the matter of the adequacy of the weld drawbead repair of the Sequoyah Unit One pressurizer relief pipe. The minority expresses its differing professional opinion in accordance with Section II.A.3.J of the memorandum, Samuel J. Chilk to William J. Dircks, dated May 1, 1980, subject, "FY 1982-86 Policy Planning and Program Guidance (PPPG)."

Non Conformance Report NCR SWP-79-S-B disclosed, that during the hot functional testing of Sequoyah Unit One, 1-RCH-93 pipe support for the pressurizer relief piping failed to slide in the vertical direction as the pressurizer expanded during heatup of the reactor coolant system. As a result the 6-inch, schedule 160 (nom. .718 wall), Type 316 stainless steel pressurizer relief pipe was bent. The related safety implicatiion was that failure of this piping could lead to an uncontrolled blowdown of the reactor coolant system.

As corrective actions, TVA had two options. The first option was to cut out the damaged pipe and replace it. This option, however, would require a system pressure test in accordance with ('77) ASME Code Section XI IWA-4400(a), which requires that after repairs by welding on the pressure retaining boundary that a system pressure test be performed. The second option was to straighten the pipe by a repair procedure which would be exempted from system hydrostatic testing. TVA, to avoid cutting out the damaged pipe, sought this exemption through IWA-4400(b)(3), which exempts from hydrostatic testing repairs by welding on the pressure retaining boundary provided that the repairs did not penetrate through the pressure boundary.

The corrective action used by TVA to straighten the pipe was the weld drawbead technique. Two 270° grooves were ground in the pipe opposite to and straddling the kink. The grooves were filled with weld metal, reground to remove that weld metal, then filled a second time with weld metal. Weld metal shrinkage provided the stressing to plastically straighten the pipe.

The repair was accepted by the Materials Engineering Branch via the memorandum, Pawlicki to Rubenstein, dated December 4, 1979, subject, "Tennessee Valley Authority, Sequoyah Nuclear Unit No. 1." TVA justified the exemption from hydrostatic testing of the system after the repair on the basis of TWA-4400(b)(3), claiming that the process of welding to realign the pipe did not result in penetration of the reactor coolant boundary. The minority challenged acceptance of the repair on the basis that more information was needed.

The memorandum, Gustafson to Pawlicki, dated January 25, 1980, subject, "Trip Report of Visit to Tennessee Valley Authority Sequoyah Nuclear Plant, Unit-1," which reported on a visit to the Sequoyah site, found the repair acceptable. The minority, after review of this memorandum and documentation related thereto, recommended in the memorandum, Halapatz to Pawlicki, dated February 27, 1980, subject, "Sequoyah Unit One Weld Drawbead Realignment of 6" Pressurizer Relief Pipe," that the Materials Engineering Branch defer acceptance of the repair pending the development and review of additional information. The minority was then advised by his assistant director that he was to personally examine the weld mockup used to qualify the repair which had been made. The memorandum, Pawlicki to Rubenstein, dated February 28, 1980, subject, "Tennessee Valley Authority, Sequoyah Nuclear Plant, Unit No. 1, Realignment of Pressurizer Relief Pipe," then reiterated acceptance of the repair and recommended that the minority meet with TVA personnel and examine metallographic samples. On March 5 and 6, 1980, the minority visited TVA at Knoxville and performed a metallurgical examination of the mockup used for the qualification of the weld drawbead realignment of the Sequoyah Unit One pressurizer relief pipe. Metallographic evidence was documented which showed that the mockup weld was fully penetrated. Full penetration of the mockup weld, which was supposed to represent the weld repair of the damaged pressurizer relief pipe, obvously did not demonstrate compliance with Section XI IWA-4400(b)(3). This finding, in itself, provided cause for denial of exemption from hydrostatic testing of TVA's weld drawbead repair of the pressurizer relief pipe which had been made. Other inconsistencies were noted between the mockup and the actual relief pipe. For example, a different material was used in the mockup. Further, while the mockup had only one weld groove, the actual relief pipe repair used two weld grooves. In addition, metallographic evidence was documented which showed through-wall sensitization to a significant degree, indicating that a potential through-wall crack propagation path existed. Since the propagation of cracks through the pipe wall is the essential concern with respect to the integrity of the reactor coolant boundary, it is the minority opinion that intergranular corrosion tests which would expose to the test environment specimens which represent the through-wall microstructure should be performed. However, only tests of ID specimen surfaces were performed.

Given that the mockup weld was fully penetrated, the minority concluded that TVA had not qualified its exemption to system hydrostatic testing.

Disclosure of the above information let to a meeting of TVA and NRC on March 13, 1980. It was agreed that TVA would perform in situ metallography to evaluate sensitization in the actual relief pipe repair and re-radiograph the repair to determine whether or not the pressure boundary had been fully

penetrated. The examination, reported in the memorandum Mills to O'Reilly, dated April 11, 1980, subject, "Sequoyah Nuclear Plant Unit 1 - Pressurizer Relief Piping Support - NCR SWP 79-S-8 - Supplemental Information" found the weld heat affected zone to be unsensitized and therefore, that sensitized base metal underlying the weld did not encroach on the pipe ID. In addition, on the basis of radiographic examination of the repair, it was concluded that the weld did not encroach on the pipe ID, i.e., did not fully penetrate the reactor coolant pressure boundary. These results were concurred in by OIE-RII in the memorandum, Murphy to Thornburg, dated April 22, 1980, subject. "RII Report No 50-327/80-12 Concerning Inspection Performed to Evaluate Repair of Sequoyah Unit 1 Pressurizer Relief Line."

The minority considers that meaningful metallurgical conclusions cannot and should not be made from Xerox reproductions of the in situ metallography, which have been made available. Given the carbon content (.052/.059%) of the pressurizer relief pipe, the minority finds it anomalous that the weld heat affected zones did not show some sensitization, since then it is inferred that the base metal at any distance from the molten weld metal essentially did not experience some time in the 800°F to 1500°F sensitization range during weld cooling.

The matter of the sensitization of austenitic stainless steels is enveloped in controversy. Arguments are made that the weld drawbead repair welds are no different than adjoining full penetrated installation welds. In the absence of identical metallurgical histories, however, this argument is tenuous. The minority notes the safety implication involved, viz., that failure of the repaired piping cannot be isolated, which as a consequence, could lead to an uncontrolled blowdown of the reactor coolant system. The minority is of the opinion that this matter be examined to a much more definitive and conclusive end. It should also be kept in mind that the environment, which will be experienced in service by the repair, will be a calculated 0.2 ppm maximum oxygen bearing steam rather than reactor coolant water containing a residual oxygen concentration during power operationn of 0.005 ppm. BWR pipe crack experience and the lack of corrosion data on the performance of sensitized austenitic stainless steel weldments in 0.2 ppm oxygen bearing steam would suggest caution in acceptance of the Sequoyah weld drawbead repair of the pressurizer relief pipe. The argument that PWR service experience has not identified a problem with pressurizer relief pipes is tenuous, because it is unknown how many, if any, operating plants include pressurizer relief pipes which have been repaired as has Sequoyah's. Given this uncertainty, which the minority feels is related to the in situ metallography performed, the more definitive laboratory examination and corrosion testing of boat samples parted from the weld drawbead repaired Sequoyah pressurizer relief pipe is proposed for consideration.

With respect to the finding that the weld repair did not full penetrate the reactor coolant boundary, it is the minority opinion that it has not been demonstrated that the radiographic technique used has the capability to develop this conclusion. While evidence that the 2T hole in an ASTM No. 12 penetrameter was visible to TVA Level III film interpreters and OIE-RII personnel may demonstrate that defects are not present, these criteria may not necessarily demonstrate the capability of the technique to discriminate in a

radiograph between sound weld metal and sound wrought base metal underlying the weld metal. The technique must be able to provide for this distinction in order to confirm whether or not the weld has fully penetrated the reactor coolant boundary. The capability of the technique could be confirmed or denied by radiographing a known fully penetrated weld and a known partially penetrated weld in the same material and observing if a distinction can be made in film density differences in the weld root area between weld metal and wrought base meta.

Given the controversy which sometimes attends the interpretation of examination results, inspection by third party is desirable. Attention is called to an NRC position stated in the memorandum, Rubenstein to Farris, dated September 12, 1979, subject, "Qualification of Inspectors, Inspection Specialists, and Inspection Agencies for Sequoyah." The Rubenstein memorandum states the NRC position that TVA institute third part inspection for the Sequoyah nuclear plant. The Rubenstein memorandum is provided as an attachment to this memorandum. The minority opinion concludes that third party inspection is required and should be implemented in the matter of the acceptance of the weld drawbead repair of the Sequoyah Unit One pressurizer relief pipe.

J. Halapatz
Materials Engineering Branch
Division of Engineering
Office of Nuclear Reactor Regulation

Enclosure:
As stated

cc: V. S. Noonan
R. L. Tedesco
A. Schwencer
C. E. Murphy, OIE-RII
A. R. Herdt, OIE-RII
R. M. Gamble
C. Stahle
P. K. Van Doorn, OIE-RII
MTEB Reading File

DISTRIBUTION

SEP 21 1979

Docket Files	bcc: NSIC
NRC PDR	TIC
Local PDR	ACRS (16)
TERA	R. Mattson
LWR-4 File	D. Eisenhut
D. Vassallo	J. P. Knight
S. Varga	L. Shao
F. Williams	S. Pawlicki
L. Rubenstein	V. Noonan
C. Stahle	R. Gamble
M. Service	H. Conrad
ELD - C. Woodhead	C. Y. Cheng
IE (3)	S. J. Bhatt
J. Halapatz	J. M. Grant
F. B. Litton	M. Hum
C. D. Sellers	M. L. Boyle

Docket Nos.: 50-327/328

Mr. H. G. Parris
Manager of Power
Tennessee Valley Authority
500A Chestnut Street Tower II
Chattanooga, Tennessee 37401

Dear Mr. Parris:

SUBJECT: QUALIFICATION OF INSPECTORS, INSPECTION SPECIALISTS, AND INSPECTION AGENCIES FOR SEQUOYAH

In Amendment No. 61 to the Sequoyah FSAR, you stated that you will provide your own independent review of the Section XI program of the ASME Boiler and Pressure Vessel Code through the TVA central office staff in Chattanooga, Tennessee. It is TVA's policy to provide its own inspection services on the basis that TVA is a Federal agency and it is not subject to State or other non-Federal inspectors.

It is our position that TVA is not exempt from any of the requirements of 10 CFR Part 50, Section 50.55a(g)(4). Therefore, we require that TVA institute the third party inspection system of the Sequoyah nuclear power plant.

A letter of compliance is requested.

Sincerely,

Original signed by!
L. S. Rubenstein, Acting Chief
Light Water Reactors Branch No. 4
Division of Project Management

cc: See next page

Record Note: Spoke to Dick Clark, DSS, who was informed of this approach to assure consistency.
9/21/79

DPM:LWR-4
MService
9/ /79

DSS:MTEB
SPawlicki
9/ /79

OFFICE	DPM:LWR-4	OELD	DPM:LWR-4			
SURNAME	CStahle/jl	CWoodhead	LRubenstein			
DATE	9/ /79	9/ /79	9/20/79			

NRC FORM 318 (9-76) NRCM 0240

★ U.S. GOVERNMENT PRINTING OFFICE: 1979-289-769

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SEP 21 1979

Docket Nos.: 50-327/328

Mr. H. G. Parris
Manager of Power
Tennessee Valley Authority
500A Chestnut Street Tower II
Chattanooga, Tennessee 37401

Dear Mr. Parris:

SUBJECT: QUALIFICATION OF INSPECTORS, INSPECTION SPECIALISTS, AND INSPECTION AGENCIES FOR SEQUOYAH

In Amendment No. 61 to the Sequoyah FSAR, you stated that you will provide your own independent review of the Section XI program of the ASME Boiler and Pressure Vessel Code through the TVA central office staff in Chattanooga, Tennessee. It is TVA's policy to provide its own inspection services on the basis that TVA is a Federal agency and it is not subject to State or other non-Federal inspectors.

It is our position that TVA is not exempt from any of the requirements of 10 CFR Part 50, Section 50.55a(g)(4). Therefore, we require that TVA institute the third party inspection system of the Sequoyah nuclear power plant.

A letter of compliance is requested.

Sincerely,

L. S. Rubenstein

L. S. Rubenstein, Acting Chief
Light Water Reactors Branch No. 4
Division of Project Management

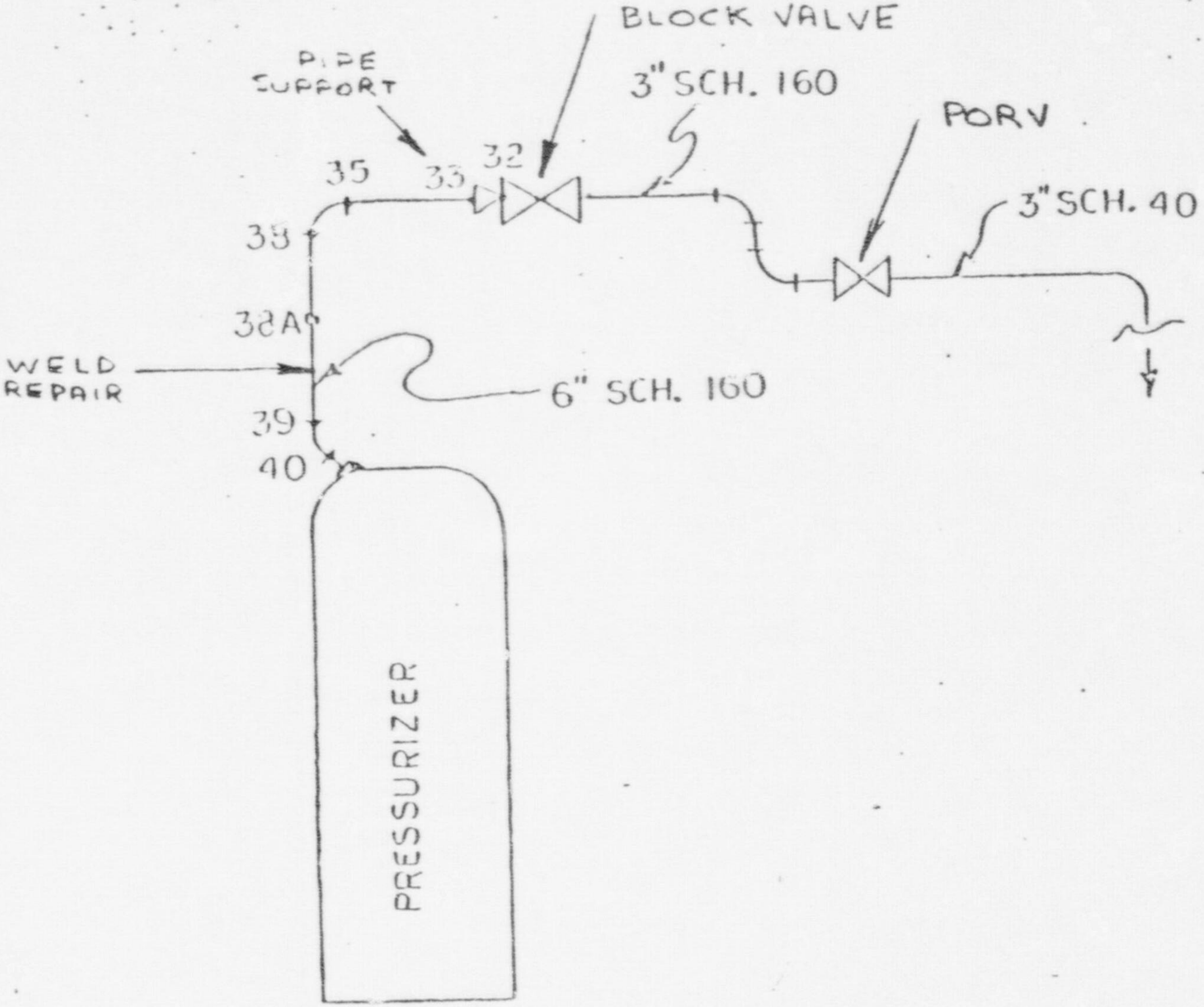
cc: See next page

ccs:
Herbert S. Sanger, Jr. Esq.
General Counsel
Tennessee Valley Authority
400 Commerce Avenue
E11B33
Knoxville, Tennessee 37902

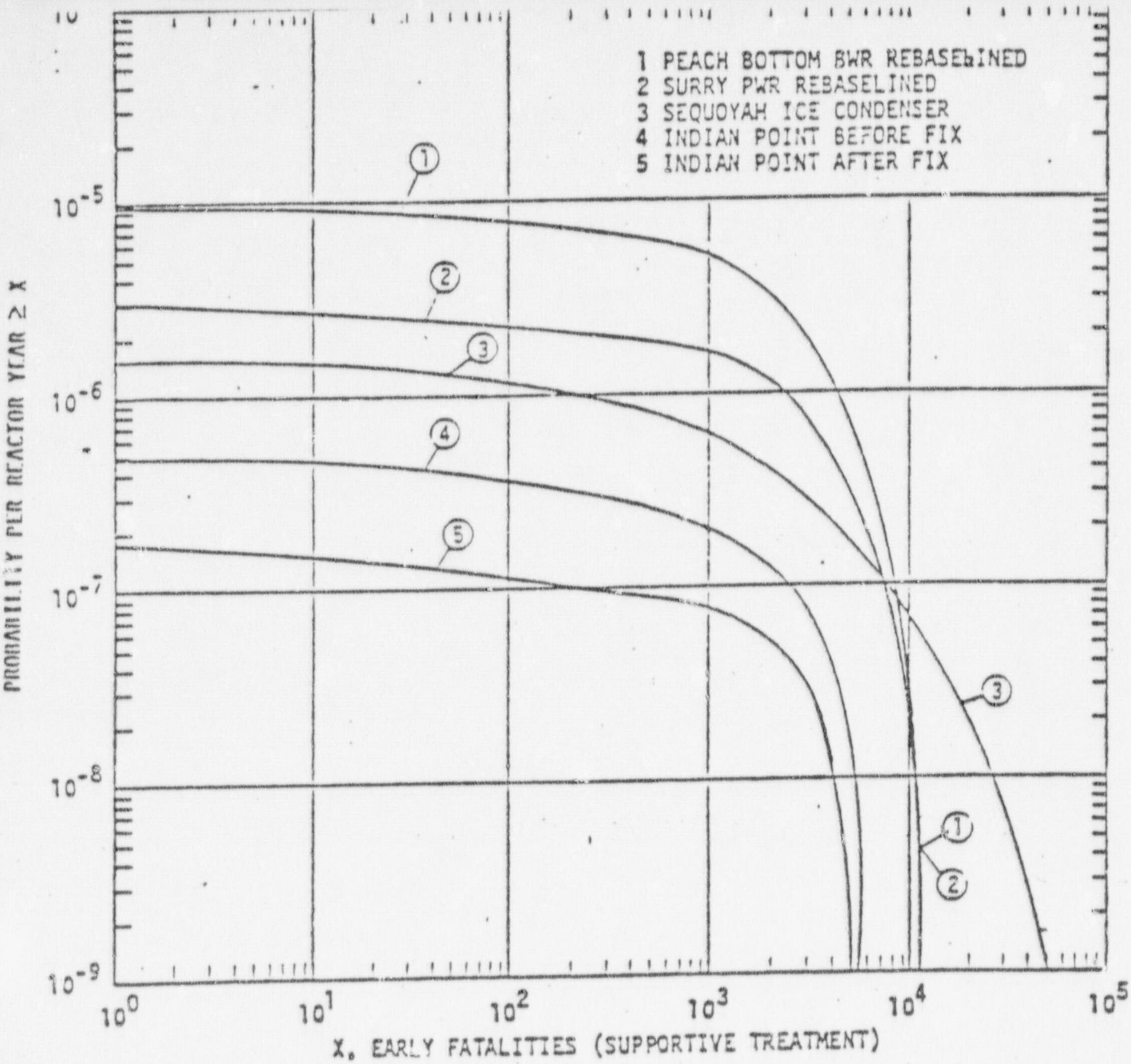
Mr. E. G. Beasley
Tennessee Valley Authority
400 Commerce Avenue
W10C131 C
Knoxville, Tennessee 37902

Mr. Michael Harding
Westinghouse Electric Corporation
P. O. Box 355
Pittsburgh, Pennsylvania 15230

Mr. David Lambert
Tennessee Valley Authority
400 Chestnut Street Tower II
Chattanooga, Tennessee 37401



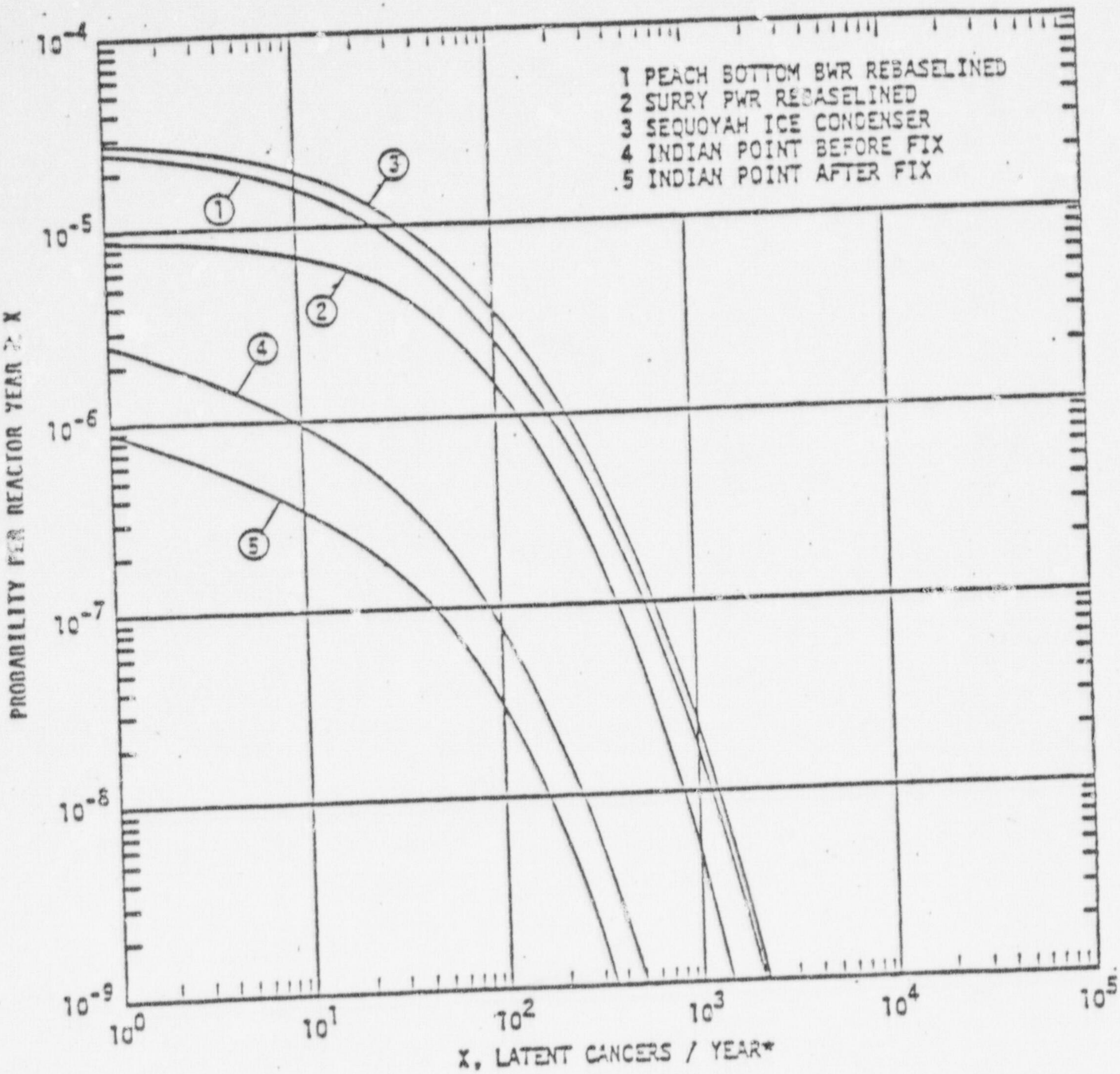
PRESSURIZER RELIEF PIPING SKETCH



NOTE: THERE ARE LARGE UNCERTAINTIES WITH THE ABSOLUTE VALUES PRESENTED IN THIS FIGURE

ASSUMPTIONS: 1) INDIAN POINT SITE
METEOROLOGY - 91 WEATHER SEQUENCES
WIND ROSE WEIGHTED 1970 CENSUS POPULATION DISTRIBUTION
UNIT 3 POWER LEVEL (3025 MWT)
2) WITHIN 10 MILES - ENTIRE CLOUD EXPOSURE + 4 HOURS GROUND EXPOSURE
NO SHIELDING
BEYOND 10 MILES - ENTIRE CLOUD EXPOSURE + 7 DAY GROUND EXPOSURE
SHIELDING BASED ON NORMAL ACTIVITY

LATENT CANCER RISK FOR DIFFERENT DESIGNS



*TOTAL LATENT CANCERS WOULD BE 30 TIMES HIGHER

NOTE: THERE ARE LARGE UNCERTAINTIES WITH THE ABSOLUTE VALUES PRESENTED IN THIS FIGURE.

ASSUMPTIONS: 1) INDIAN POINT SITE
METEOROLOGY - 91 WEATHER SEQUENCES
WIND ROSE WEIGHTED 1970 CENSUS POPULATION DISTRIBUTION
UNIT 3 POWER LEVEL (3025 MWT)

2) WITHIN 10 MILES - ENTIRE CLOUD EXPOSURE + 4 HOURS GROUND EXPOSURE
NO SHIELDING

BEYOND 10 MILES - ENTIRE CLOUD EXPOSURE + 7 DAY GROUND EXPOSURE
SHIELDING BASED ON NORMAL ACTIVITY