BETH-5 (3-65) U. S. ATOMIC ENERGY COMMISSION

APPLICANT: United Nuclear Corporation
Research and Engineering Center
Grasslands Road
Elmsford, New York 10523

DOCKET NO. 50-290

Proof Test Facility

DOCUMENTS

DATE DOCKETED	DESCRIPTION
May 5, 1967	Ltr. 4-28-67 fm UNC submitted as an application for a Class 104 Utilization Lic for the Proof Test Facility, which is a light water moderated, zero-power critical assembly and will be used for the purpose of providing a final reactivity test on manufactured fuel assemblies; the PTF is located in the Critical Facility building.....about 4 miles west of Pawling, NY. Included: UNC-5175 --"Proof Test Facility (PTF) Safety Analysis Report," dtd 4-28-67. (22 cys of ltr rec'd and 40 cys of report rec'd --not notarized) #1476
May 16, 1967	Ltr 5-16-67 to UNC trans Notice of Receipt of Appl for Const. Permit and Facility license....
May 17, 1967	Ltr 5-17-67 to OFR trans Notice of Receipt of Appl for Const. permit & Facility license for publication.
May 17, 1967	Ltr 5-17-67 to Townsend trans the following: Federal Register Notice Appl. dtd 4-28-67, for UNC.
May 19, 1967	Ltr 5-19-67 to UNC requesting addtl info to support appl ref. meeting 5-17-67.
May 25, 1967	Notice of Receipt of appl for Const. Permit & Facility license filed W/FR 5-22-67; published 5-23-67; Citation 32 F.R. 7541; AD- None

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BETH-5 (3-65) U. S. ATOMIC ENERGY COMMISSION

APPLICANT United Nuclear Corp
Research & Engineering Center DOCKET NO. 50-290

Proof Testing Facility

DOCUMENTS

DATE DOCKETED	DESCRIPTION
June 12, 1967	Ltr 6-9-67 fm UNC submitted as AMDT # 1 to appl. ack our ltr of 5-19-67 & trans the following: UNC- Consolidated Balance Sheet Fig.1.1- "Scram Circuit Schematic Diagram" Fig.7.1- "Ion Chamber Support Structure" Drawings # 17267, 17268, 17271 and 17257 (1cy ltr & 22 cys of encl rec'd.....notarized 6-9-67...#1957 (Note: Mirerard Ref. markings on Dwgings (Per Fleury 6-15-67)
June 14, 1967	Ltr 6-14-67 to S/C (HEW) Rockville trans the following: UNC, appl dtd 4-28-67 Ltr 6-9-67, submitted as Amdt # 1 to appl.
June 14, 1967	Ltr 6-14-67 to Townsend trans UNC ltr dtd 6-9-67, submitted Amdt # 1
June 14, 1967	Ltr 6-14-67 to David Schoentag, Supervisor, Town of Beekman, Poughquag, N.Y. trans Notice of Receipt of Apol for Const Permit & facility license.....
July 10, 1967	Ltr dtd 7-7-67 fm UNC trans as per telecon cy of United Nuclear Corporation 1967 Annual Report..(Ltr 1 orig-encl. 3 cys) #2248
July 17, 1967	Ltr 7-17-67 to UNC trans the following: Notice of Proposed Issuance of a Const. Permit. Federal Register Notice Safety Evaluation Report, (AEC)

Continued)

BETH-5 (3-65) U. S. ATOMIC ENERGY COMMISSION

APPLICANT United Nuclear Corporation DOCKET NO. 50-290

Research & Engineering Center

Proof Testing Facility

DOCUMENTS

DATE DOCKETED	DESCRIPTION
July 17, 1967	Ltr 7-17-67 to the following and trans Notice of Proposed Issuance of a Const. Permit and Safety Evaluation Report: Schoentag, Supervisor. Town of Beekman, N.Y. Townsend, Dir. Office Atomic Space Development. SO/ PHS (HEW), Rockville
July 17, 1967	Ltr dtd 7-17-67 to OFR trans : NOTICE OF PROPOSED ISSUANCE OF CONSTRUCTION PERMIT....
July 20, 1967	Ltr 7-20-67 to JCAE trans the following: Notice of proposed issuance of Const. permit Safety Evaluation (AEC) dtd 7-14-67
July 21, 1967	Notice of proposed issuance of const. permit, Citation: 32 F.R. 10617 Dtd filed: 7-18-67; Dtd published: 7-19-67; AD 8-4-67; (15 days after publication)
July 28, 1967	Ltr dtd 7-26-67 fm D. C. Schoentag, Board of Supervisors, Poughkeepsie, N. Y. req date of proposed issuance of const. Permit in FR in case Town Board decides to req hearing or to intervene & to advise Board in layman's language operation & scope of facility (1 cy only) #2493
August 2, 1967	Ltr, 8-2-67 , to David C. Schoentag, Supervisor, Town of Beekman giving info req in ltr of 7-26-67 & advising deadline if town wishes to petition to intervene.....
August 7, 1967	Ltr, 8-7-67, to United Nuclear trans the following: 1. Construction Permit No. CPCX-27 , dtd 8-4-67 2. Notice of Issuance

Continued

BETH-5 (3-65) U. S. ATOMIC ENERGY COMMISSION

APPLICANT United Nuclear Corporation DOCKET NO. 50-290
Research & Engineering Center
Proof Testing Facility

DOCUMENTS

DATE DOCKETED	DESCRIPTION
August 7, 1967	Ltr, 8-7-67, to David Schoentag, Super. Town of Beekman, to inform him of issuance of Construction Permit to United Nuclear & trans. cy of Construction Permit No. CPCX-27 & Federal Reg. Notice,...
August 7, 1967	Ltr, 8-7-67, trans Notice of Issuance and Construction Permit to the following: SG/HEW, Rockville, Md. Oliver Townsend, Dir. Off. of A&SD
August 10, 1967	Ltr, 8-10-67, to OFR trans Notice of Issuance of Construction Permit, dtd8-4-67 for earliest publication...
August 17, 1967	Notice of Issuance of Construction Permit, filed w/FR 8-15-67; Published 8-16-67; Citation 32 F. R. 11813; AD-None...
August 30, 1967	Ltr, 8-30-67, to United Nuclear trans the following: Draft- Technical Specifications for United Nuclear which reflect comments to draft Specs submitted on 4-28-67 by them ...
Sept. 25, 1967	Ltr, 9-25-67, to United Nuclear informing them latest completion date is 11-1-67 and if extension is needed amendment is necessary and reason for extension of time ...
Sept. 27, 1967	Ltr, 9-25-67, in reply to ours of 8-30-67 trans comments on draft Tech Specs and including : Figure 1-Scram Circuit Diagram (1 orig rec'd.....#3194)
Continued	

BETH-5 (3-65) U. S. ATOMIC ENERGY COMMISSION

APPLICANT United Nuclear Corporation
Research & Engineering Center
Proof Testing Facility

DOCKET NO. 50-290

DOCUMENTS

DATE DOCKETED	DESCRIPTION
Oct. 5, 1967	Ltr, 10-3-67, fm United Nuclear ref ours of 9-25-67 advising that they do not anticipate any problems in meeting completion date, but will let us know if the unexpected does occur... (1 cy rec'd.....#3299)
Oct. 16, 1967	Ltr, 10-13-67, fm United Nuclear requesting extension of construction permit No. CPCX-27 which expires 11-1-67 until 12-1-67 due to delay as a result of steel workers strike. Notarized 10-13-67 (1 orig rec'd.....#3400)
Oct 23, 1967	Ltr 10-23-67 to UNC auth extension of const permit until 12-1-67 Included: 1. Order of extension & 2. Federal Register Notice.
Oct 26, 1967	Ltr 10-26-67 to OFR trans notice of extension of completion dtd. for publication.
Nov. 1, 1967	Ltr 10-30-67 fm UNC submitted as AMDT # 2 to their 9-28-67 ltr appl to include those portions of their 9-25-67 ltr whichare not being revised by present communication...& trans: Figure 1- Scram Circuit Diagram, replaces Fig 1-1 of Amdt # 1 (3 signed cys & 19 conf'd cys of Amdt # 2 & 22 cys of Fig 1 rec'd.....notarized 10-30-67.....#3691
Nov. 2, 1967	Ltr 11-2-67 to Schoentag Supervisor of Town of Beekman, N.Y. advising of extension of const. permit, included: Or r & Fed Reg Notice.
Nov. 2, 1967	Ltr 11-2-67 to the following SG/PHS (HEW) Rockville, & Townsend Dir. Atomic & Space Development trans the following: 1. Order extension completion date 2. Notice of extension of completion date 3. UNC ltr dtd 10-13-67
Nov 2, 1967 (Continued)	Notice of extension of completion date (DRL order issued 12-23-67, extending completion date until 12-1-67, Citation: 32 F.R. 15130; Dtd filed: 10-31-67; Dtd published 11-1-67; AD: None

RETH-5 (3-65) U. S. ATOMIC ENERGY COMMISSION

United Nuclear Corporation	50-290
APPLICANT	DOCKET NO.
Research & Engineering Center Proof Test Facility	

DOCUMENTS

DATE DOCKETED	DESCRIPTION
Nov. 8, 1967	Ltr, 11-8-67, (Mailed 11-14-67), to United Nuclear trans the following: Notice of Proposed Issuance of Facility License filed w/FR Safety Evaluation , dtd 11-8-67
Nov. 14, 1967	Ltr, 11-14-67, trans (1) Federal Register Notice, dtd 11-8-67, (2) Safety Evaluation & (3) UNC's amendments dtd 9-25-67 & 10-30-67 to the following: Oliver Towsend, Off. of A&SD SG/HEW, Rockville, Md.
Nov. 14, 1967	Ltr, 11-14-67, to David Schoentag, Super. Town of Beekman, trans the following: Federal Register Notice Safety Evaluction
Nov. 14, 1967	Ltr, 11-14-67, to OFR trans Notice of Proposed Issuance of Facility License to be published at earliest possible date.
Nov. 17, 1967	Notice of Proposed Issuance of Fac. lic. Citation: 32 F.R. 15770 Dtd filed 11-15-67; Dtd Published: 11-16-67, AD: None
Dec. 2, 1967	Ltr, 12-2-67, to UNC trans: License No. CX-25- Authorizing possession , use and operation of Proof Test Facility w/APPENDIX A Federal Register Notice Amendment No. 6 to Indemnity Agreement No. B-16- to be signed and returned.
Dec 5, 1967	Ltr 12-5-67 to OFR requesting publication of Notice of Issuance of Facility License. # CX-25.
Dec 5, 1967	Ltr 12-5-67 to David Schoentag, Super Town of Beekman, N.Y. advising of the issuance of lic CX-25....encl. cy of lic CX-25 & Fed Reg. Notice..
(Continued)	

BETN-5 (3-65) U. S. ATOMIC ENERGY COMMISSION

United Nuclear Corp.	50-290
APPLICANT	DOCKET NO.
Research & Engring Center Proof Test Facility	

DOCUMENTS

DATE DOCKETED	DESCRIPTION
Dec 5, 1967	Ltr 12-5-67 to GSO/ PHS (HEW) trans cy of lic. CX-25 & Fed. Reg. Notice.
Dec. 5, 1967	Ltr 12-5-67 to Oliver Townsend trans cys of lic CX-25 & Fed Reg. Notice.
Dec. 11, 1967	Notice of Issuance of Facility License CX-25, Citation 32 F.R. 17632; Dtd filed: 12-8-67; Dtd Published 12-9-67; AD;None
May 6, 1968	Ltr, dtd 4-29-68, fm United Nuclear Corporation, reporting the measurement of a small positive moderator temperature coefficient at room temperature.....in accordance w/ req. of Lic CX-25..... (1 cy rec'd.....#1611)
Sept 11, 1968 IP	Ltr, dtd 9-10-68, fm United Nuclear Corporation requesting authority to amend CX-25 as described in the following document: UNC-5220 requesting permission to use plutonium-bearing fuel assemblies in the Proof Test Facility (20 Notarized cys rec'd of each.....#3181)
Nov 21, 1968	Ltr, dtd 11-19-68, fm United Nuclear trans the following as requested in telcon between Mr. Berger, AEC and Mr. Tomonto, UNC in October 1968: SUPPLEMENT A OF AMENDMENT 1 TO FACILITY LICENSE CX-25 FOR THE RPOOP TEST FACILITY Subject: Additional Analysis of the Proof Test Facility Design Accident for a Core Containing All PuO_2 -UO_2 Fuel Rods, dtd 11-15-68 NDEO -1726 (1 Orig ltr.....6 cys Suppl A rec'd.....#5015)
(cont'd)	

FOR INTERNAL USE ONLY

BETH-5 (3-65) U. S. ATOMIC ENERGY COMMISSION

APPLICANT: United Nuclear Corporation DOCKET NO. 50-290

DATE DOCKETED	DESCRIPTION OF DOCUMENTS
Dec 2, 1968	Ltr, dtd 12-2-68, to United Nuclear in ref to Safeguards inspection of the special nuclear material.
Jan 21, 1969	Ltr, dtd 1-13-69, fm United Nuclear advising that 24 kilograms of plutonium will be associated w/CX-25....the amount having been inadvertantly omitted fm previous submission. (1 Orig ltr rec'd....#163)
Feb 5, 1969	Ltr, dtd 2-5-69, to United Nuclear enclosing: AMENDMENT NO. 1 to CX-25, authorizing UNC to receive possess and use up to 24 kilograms of plutonium Federal Register Notice Safety Evaluation
Jun 24, 1970	Ltr 6-18-70 fm UNC trans the following reply to our 6-8-70 ltr: Endorsements Nos 1 thru 7 and 62 thru 64 to Policy No. NF-3 (1 cy ltr & 1 cy encl rec'd.....#1853)
June 30, 1971	Ltr 6-29-71 fm UNC notarized adv Gulf Oil & UNC will form a joint venture corporation commencing business as GUNFC and trans: List of Principal Officers ATCHMT "A" Partial Organization Chart ATCHMT "B" Resume of Arnold R. Fritsch ATCHMT "C" Resume of Kenneth L. Wiley 1970 Annual Fin Rpts for UNC & Gulf OIL (3 signed & 19 conf'd, 22 cys encl rec'd & 7 cys Annual Rpt...#2936)
Continued	

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BETH-5 (3-65) U. S. ATOMIC ENERGY COMMISSION

APPLICANT: United Nuclear Corporation DOCKET NO. 50-290

DATE DOCKETED	DESCRIPTION OF DOCUMENTS
July 6, 1971	Ltr 7-1-71 fm UNC req trans of Lic R-49 (50-101) & CX-25 (50-290) .....be trans to Gulf United Nuclear Fuel Corp (3 signed & 19 conf'd cys.....#3040)
July 6, 1971	Ltr 7-1-71 fm GUNFC req trans of lic R-49 (50-101) & CX-25 (50-290) in accordance with UNC ltr dtd 6-29-71 (3 Orig & 19 conf'd cys..... .....#3049..)
Aug. 2, 1971	Ltr 7-22-71 fm UNC notarized 7-22-71 re their 6-29-71 & 7-1-71 req...furn addl info in support of req & trans the following: (1) Gulf United Nuclear Fuels Corp Pro-Program Balance Sheet as of 3-31-71 (2) Lease Agreement w/attchmt's Exhibit A & B (3 signed & 19 conf'd, 22 cys as encl rec'd.....#3434)
Feb 17, 1972	Ltr 2-8-72 fm GUNFC notarized 2-8-72....request joint appl of GUNFC & UNC be treated as a Class 104 License (3 signed notarized & 19 conf'd.....#6000)
Apr 25, 1972	Ltr 4-25-72 to UNC trans Amdt Nos. 2 & 7; Amdt #11 to Indemnity Agreement No. B-16 and Federal Register Notice
Nov. 9, 1972 ***	Ltr 11-6-72 from Gulf United Nuclear Fuels Corp. adv that Mr. Robert E. Kopp is responsibilibe for all correspondence & req all communcations be directed to Mr. Kopp ..... (8 conf'd cys of ltr rec'd.....#6167-A)
Apr 18, 1973 ***	Ltr 4-18-73 to UNC trans for info cy of preamble for the new consolidated series of "Regulatory Guides" & cy of list of guides under development

APPLICANT: UNITED NUCLEAR CORPORATION DOCKET NO. 50-290

DATE DOCKETED	DESCRIPTION OF DOCUMENTS
Aug 6, 1973 ***	Ltr 8-2-73 fm GUNC req an Amdt to Lic CX-25. (3 signed orig.....22 cys rec'd.....#5994)
Aug 23, 1973 ***	Ltr 8-23-73 to UNC adv that AEC has amended its regulations to require operators who control nuclear power ~~plants~~ reactors to complete a re-qualification program & trans Amdt to CFR Part 50 & Part 55
Oct 25, 1973 ***	Ltr 10-23-73 fm GUNFC notarized 10-23-73 adv AEC of their intent to terminate licenses R-49 & CX-25 & to dismantle & decommission both the PLATR & PTF w/attached tables 1 thru 3. (3 signed orig.....39 cys rec'd.....#7811)
Nov 5, 1973 ***	Ltr 11-1-73 fm GUNFC notarized 11-1-73 re their 10-23-73 Appl furnishing addl info re Pawling Test Rig (PLATR) & Pawling Test Facility (PTF) Dismantling Plan. (4 signed orig.....39 cys rec'd.....#8009)
Nov 21, 1973 ***	Ltr 11-19-73 fm Gen Atomic Co notarized 11-19-73 req written consent to acquire by transfer fm Gulf Oil all rights, title & interest, whether direct or indirect of Gulf Oil in lic or appl for lic & trans..... (1)APP "A"- lic or appl for lic which are held or have been made in name of Gulf or div of (GZES, GCAC, GUNF & AGNS) (2) APP "B"- info pertinent to ident of Appl & General partners. (3 signed orig.....28 cys rec'd.....#8385)
Nov 30, 1973 ***	Ltr 11-30-73 to UNC regarding ltr dtd 11-6-73 concerning Admts to 10 CFR Parts 50, 70, and 73 and trans Admts thereto dtd 11-6-73 and request submittal of physical security plan.
Dec 5, 1973 ***	Ltr 11-30-73 fm UNC re ltr 10-23-73 fm GUNFC trans cy of ltr dtd 10-23-73 fm GUNFC to Skovholt adv AEC of plans to Dismantle & Decommission both PLATR & PTF. (23 signed orig.....23 cys rec'd.....#8655)

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***DENOTES SENT TO PDR
****DENOTES SENT TO LPDR

BETH-5 (3-65) U. S. ATOMIC ENERGY COMMISSION PAGE: 11

APPLICANT: UNITED NUCLEAR CORPORATION DOCKET NO. 50-290

DATE RECEIVED	DESCRIPTION OF DOCUMENTS
Mar 8, 1974	Ltr 3-8-74 to UNC trans Amdt to Indemnity Agreement No. B-16, Amdt #12 & Amdt to 10 CFR Part 140
Mar 27, 1974 ***	Ltr 3-21-74 fm UNC trans Amdt to Indemnity Agreement No. B-16, Amdt #12 returned, signed & accepted, dtd: 3-20-74 (1 signed.....1 cy rec'd.....#2598)
Mar 29, 1974 ***	Ltr notarized 3-13-74 fm GNFC requesting termination of Lic No. R-49 & CX-25...trans Results of Radiological survey of critical facility Bldg - PLATR & PTF...as per our ltr 1-25-74 (3 signed.....20 cys rec'd.....#2331)
June 25, 1974 ***	Ltr 6-25-74 to GUNC trans Lic. Termination Order & Amdt #13 to Indemnity Agreement No B-16.
Mar 21, 1975 ***	Ltr 3-21-75 to UNC trans (1) Amdt to Indemnity Agreement, (2) Amdt to 10 CFR Part 140
May 2, 1975	Ltr 5-2-75 to GNFC trans Amdt No. 13 to Indemnity Agreement No. B-16

FOR: Low power (5 watt) heavy water moderated and reflected reactor
(Pawling Research Reactor)
SITE: NDA's Critical Facility Building, located at Pawling, N. Y.

FOR INTERNAL USE ONLY

BETH-5 (3-65) U. S. ATOMIC ENERGY COMMISSION Page 1

APPLICANT: Nuclear Development Corporation of America DOCKET NO. 50-101
5 New Street, White Plains, N.Y.

DATE DOCKETED	DESCRIPTION OF DOCUMENTS
April 7, 1958	Ltr. 4/2/58 from NDA (Arnold M. Zais) applying for const. permit and license to cover const. & operation of a low power (5 watt) heavy w. moderated and reflected reactor to be known as Pawling Research Reac to be located in NDA's Critical Facility Building located at Pawlin N. Y., for sale to universities, medical centers and other research institutions. Application included: Report NDA 3002-1 - Pawling Research Reactor Hazards Summary Rep dtd. 2/28/58. Report NDA 2-74 - Operating Manual for the NDA Pawling Critical Facility, dtd. 2/28/58. (2 cys. ltr. & 5 cys. of each report rec'd--notarized 4/2/58).
April 21, 1958	Ltr. 4/21/58 to Gov. Harriman (New York) informing him of receipt o appl. by NDA, etc.
May 29, 1958	Ltr. 5/29/58 to NDA (Mr. Zais) informing them that appl. of 4/2 has been assigned a new Docket No. inasmuch as License DA-5 (Secret was iss. in connec. with the performance of critical experiments an is not deemed proper to amend it to auth. const. & operation of and conduct of research..... . Req. addl. info re. earliest and lates completion dates, period of license, etc.
July 7, 1958	Ltr. 6/30/58 from NDA furnishing addl. hazards, financial, etc. inf requested 5/29/58. Included: 10th Annual Report, dated March 31, 1958. (1 cy. rec'd 7/1 & 5 a cys. rec'd 7/7/58). (2 cys. of ltr. rec'd 7/1/58 & 13 addl. cys. rec'd 7/7/58) (Ltr dtd 6/30/58). (Ltr. dtd. 7/3/58 from NDA furnished addl. cy. requested by phone 7/2/58--13 cys. of 6/30/58 ltr.; 5 cys. Annual Report; 13 cys. of NDA 3002-1 and 10 copies of NDA 2-74).
July 11, 1958	Ltr. 7/11/58 to NDA ack. receipt of 7/3 & 6/30/58 ltrs.

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U. S. ATOMIC ENERGY COMMISSION

APPLICANT	Nuclear Development Corporation of America 5 New Street, White Plains, N. Y.
	DOCKET NO. [illegible]
	ION: Low power (5 watt) heavy water moderated and reflected reactor (Pawling Research Reactor)
	SITE: NDA's Critical Facility Building, located at Pawling, N. Y.

DOCUMENTS

DATE DOCKETED	DESCRIPTION
August 7, 1958	Ltr. 8/5/58 from NDA advising us of their understanding of [illegible] necessary for issuance of const. permit, etc. (2 cys. rec'd (Suppl. only)
August 18, 1958	Ltr. 8/12/58 from NDA trans. financial statements requested [illegible] telephone:
	Schedule of Profit and Loss for the Year ended 3/31/57,
	Statement of Retained Earnings as at March 31, 1957, and
	Consolidated Balance Sheet as at March 31, 1957. (2 cys. ltr. & 7 cys. of ea. statement rec'd 8/21/58--not [illegible]
August 27, 1958	Ltr. 8/25/58 from NDA (Peter C. Murphy) trans:
August 27, 1958	"Certificate of Nuclear Energy Liability Insurance", dtd. 4/1/58, Binder No. 5 issued to NDA on behalf of [illegible] (2 cys. ltr. & 5 cys. of Binder rec'd--eff. 1/1/58 to [illegible]
August 29, 1958	Ltr. 8/29/58 to NDA ack. receipt of NELIA binder.
September 16, 1958	Ltr. 9/16/58 to NDA trans:
September 16, 1958	"Notice of Proposed Issuance of Construction Permit and Facility License to Nuclear Development Corporation of America", dtd. 9/16/58, and
September 16, 1958	Memorandum, dtd. 9/16/58.
September 16, 1958	Ltr. 9/16/58 to Gov. Harriman (N.J.) trans. copy of notice & memo of 9/16/58 for NDA.
October 7, 1958	Ltr. 10/7/58 to NDA trans:
October 7, 1958	Construction Permit No. CPRR-29 issued to NDA 10/7/58 to auth. const. of low power research reactor.
October 7, 1958	"Notice of Issuance of Construction Permit" to NDA, dtd. 10/7/58, filed with FR.
October 7, 1958	Ltr. 10/7/58 to Gov. Harriman (NY) trans. copy of [illegible] permit.

U. S. ATOMIC ENERGY COMMISSION

APPLICANT	Nuclear Development Corporation of America 5 New Street, White Plains, N. Y.	DOCKET NO. 50-[illegible]
	FOR: Low power (5 watt) heavey water-moderated and -reflected react[illegible] (Pawling Research Reactor) SITE: NDA's Critical Facility Building, located at Pawling, N. Y.	

DOCUMENTS

DATE DOCKETED	DESCRIPTION
October 22, 1958	Ltr. 10/22/58 to NDA trans:
October 22, 1958	Facility License No. R-49 issued to NDA 10/22/58 for oper[illegible] of "Pawling Reactor".
October 22, 1958	"Notice of Issuance of Facility License" to NDA, dtd. 10/22/[illegible] to FR.
October 22, 1958	Ltr. 10/22/58 to Gov. Harriman (NY) trans. copy of Lic. R-[illegible]
November 26, 1958	Ltr. 11/26/58 to NDA trans. two copies of interim indemnity agreement for sign., date & return of one copy to us--re Lic. R-49.
December 8, 1958	Ltr. 12/8/58 from NDA returning a signed copy of the interim indemnity agreement -- sgd. "Arnold M. Zais, Treasurer" & Accepted "12/5/58".
February 26, 1960	Ltr. 1-29-60 from NDA to LJ re. lattice experiments they pro[illegible] and their being license exempted, etc.--suppl. only.
February 26, 1960	Telegram 2-18-60 from NDA requesting amendment to License R-[illegible] to auth. performance of experiments described in Hazards [illegible] Report NDA-2109-3, dtd. 9-30-59, -- "Pawling Lattice Test [illegible] Advising us that cys. of HSReport sent to NYOO & [illegible]; [illegible] (PLATR)
February 26, 1960	Report NDA-2109-3--"Pawling Lattice Test Rig Hazards [illegible] Report", dtd. 9-30-59 (6 cys. rec'd in Dec. 59 fm NYOO). (1 p addl. cys. rec'd 3-16-60)
February 26, 1960	Ltr. 2-19-60 from NDA formally requesting amendment to Lic. [illegible] to auth performance of experiments in NDA-2109-3 in the Pawli[illegible] Research Reactor. (14 cys. rec'd 2-23-60--not. 2-19-[illegible]).
February 26, 1960	Ltr. 2-25-60 to NDA trans: Amendment No. 1 to License No. R-49, dtd. 2-[illegible], [illegible] install. of a new heavy water moderated, graphite and heav[illegible] ..., designated as Pawling Lattice Rig, in its Pawling Research Reactor and to conduct exps. at powerlevels [illegible] 50 watts... AND the receipt, possession & use of [illegible]

U. S. ATOMIC ENERGY COMMISSION

APPLICANT	Nuclear Development Corporation of America 5 New Street, White Plains, N. Y.	DOCKET NO. 50-101
FOR:	Low Power (5 watt) heavy water-moderated and -reflected reactor (Pawling Research Reactor) & Pawling Lattice Test Rig (PLATR).	
SITE:	NDA's Critical Facility Bldg., located at Pawling,	

DOCUMENTS

DATE DOCKETED	DESCRIPTION
Feb. 26, 1960	Ltr. 2-25 to NDA trans continued: "Notice of Issuance of Facility License Amendment" to NDA, dtd. 2-25-60, and Hazards Analysis, dtd. 2-25-60,
February 25, 1960	Ltr. 2-25-60 to Gov. Rockefeller (NY) trans. cy. of NDA's notice & of 2-25-60.
April 15, 1960	Ltr. 4-15-60 to Nuclear Development Corp trans. cys. of [illegible], to Part 140, etc., and advising them that they will be required to furnish financial protection in amt. of $1,000,000, prior to [illegible].
May 6, 1960	Tel 5-6-60 to NDA advising them that reply to our 4-15-60 ltr. req by close of business 5-9-60.
May 9, 1960	Ltr. 5-6-60 from NDA furnishing: Declarations Page to Policy NF-3, Endorsement 8 to Policy NF-3--which is for $1,000,000, Declarations to Policy NF-5, and Endorsement 9 to Policy NF-5--which is for $1,000,000. (1 cy each of ltr. & encls. rec'd).
June 3, 1960	Ltr. 6-3-60 to NDA advising them that proof of financial protection under License R-49 is satisfactory.
Jan. 10, 1961	Telegram 1-10-61 to NDA advising them that we are surveying all lic reactor facilities for info gained from oper exp. which indicates confirm. of or variance from nuclear character. shown in lic. appl time of lic issuance; accordingly, a report on max excess reactivity, total control rod worth, etc., should be filed within 20 days.
Feb. 1, 1961	Ltr. 1-26-61 from NDA replying to our telegram of 1-10-61. Includes Attachment RH -1765A. (15 complete cys. rec'd--all reproduced sim
Feb. 3, 1961	Ltr. 2-3-61 to NDA ack. 1-26-61 filing.

U.S. ATOMIC ENERGY COMMISSION

APPLICANT: Nuclear Development Corporation of America, 5 New Street, White Plains, N.Y. DOCKET NO. 50-101

FCR: Low Power (5watt) heavy water-moderated and -reflected reactor (Pawling Research Reactor) & Pawling Lattice Test Rig (PLATR).

SITE: NDA's Critical Facility Bldg., located at Pawling, N. Y.

DOCUMENTS

DATE DOCKETED	DESCRIPTION
Apr 5, 1961	Ltr. 4-5-61 to NDA req. info as to current status of and future plans for the operating organizations for each reactor facility by 4-30-61.
May 2, 1961	Ltr. 5-1-61 from NDA replying to our ltr. of 4-5-61 re operating organization. Included: "Job Descriptions". (15 cys. rec'd between this case & 50-99--5/6
May 9, 1961	Ltr. 5-1-61 from NDA advising that assets and business of NDA will be transfer to United Nuclear Corporation on or about 5-31-61, with all facilities and personnel and practices and procedures in effect as before and requesting that Facil License R-49 be transferred to United Nuclear Corporation. Included: Attachment A -- Nuclear Development Corporation of America Notice of Special Meeting of Stockholders--for. 5-24-61, dtd. 5-2-61. (2 cys of ltr. & attach. rec'd 5-5-61--not notarized). (25 addl. cys. rec'd betwe this case & 40, 70 & 90 cases on 5-10-61 & given to LJ).
May 10, 1961	Ltr. 5-9-61 from NDA advising us that they plan to establish UNC's headquarter office in NY City in the near future, but in the meantime the Chairman and Pr of UNC (Mr. William C. Foster) can be reached in D. C. FE 3-2375 and mail addressed to Foster c/o Olin Mathieson Nuclear Corp at 460 Park Avenue, NY 22 and confirming divisional designations in transfer of licenses. (1 cy rec'd & 30 cys. verifaxed -- 22 & original to LJ).
May 18, 1961	Ltr. 5-12-61 from NDA requesting that Lic. CX-8 under 50-23 be terminated inas as Lic. R-49 superseded CX-8. -- suppl. only in this case--(See 50-23).
May 22, '61	Ltr. 5-22-61 to NDA ack. 5-1 & 5-9 re. re transfer to United Nuclear Corp.
May 23, '61	Ltr. 5-23-61 to NDA trans. cy of notice of amend. to Part 140 advising that as licensee who has furnished nuclear energy liability ins. as proof of financial protection and has entered into an IIA w/AEC, it will be necessary that the IIA be replaced by a definitive agreement as provided in amend. to Part 140, etc.
May 24, '61	Ltr. 5-23-61 from NDA trans. the following report and requesting that Lic. R-49 be amended to auth performance of PLATR critical exps. under the conditions of the HS Report NDA-2109-3 as modified by Amend. No. 1 to NDA 2109-3 (NDA-2131-37): Report NDA-2131-37--"Amendment I to Pawling Lattice Test RIG Hazards Summary Report" (NDA-2109-3), dtd. 5-19-61, which brings info submitted in NDA-2109-3 up to date. (14 cys. rec'd of ltr. & report--not notarized).

U.S. ATOMIC ENERGY COMMISSION

APPLICANT - United Nuclear Corporation — DOCKET NO. 50-[illegible]
5 New Street, White Plains, N.Y.

FOR: Low Power (5 watt) heavy water-moderated and -reflected reactor (Training Research Reactor) & Pawling Lattice Test Rig (PLATR).

SITE: NDA's Critical Facility Bldg., located at Pawling, N.Y.

DOCUMENTS

DATE DOCKETED	DESCRIPTION
May 26, 1961	Ltr. 5-26-61 to ACRS trans 3 cys. of NDA's 5-23-61 filing.
June 7, 1961	Ltr. 6-6-61 to NDA trans: "Questionnaire on Gaseous Effluent Practices" for t reply by 7-31-61.
June 12, '61	Ltr. 6-8-61 from Marsh & McLennan trans: Endorsement 11 to NF-3, eff. 6-1-61, which changes name of insured to United Nuclear Corporation. (1 cy ltr. & 8 cys of end. rec'd). — Ltr 6-26-61 from UNC requesting copy of indemnity agreement.
Jun 27, 61	Ltr. 6-27-61 to UNC requesting reply to our 5-23-61 ltr. on Part 110 re propose description of the location for consideration in connection with the preparation of a definitive agreement.
y 6, 61	Ltr. 7-5-61 from United Nuclear req. cy of indemnity agreement for NDA for Lic. R-49. -- supple. only.
July 12, 61	Ltr. 7-10-61 from United Nuclear Corp. replying to our 6-6-61 questionnaire to U stating that operation of their reactor does not create a gaseous radioactive effluent problem... (1 cy rec'd).
July 13, 61	Ltr. 7-13-61 to United Nuclear Corp. trans: Amendment No. 2 to License No. R-49, dtd. 7-13-61, which changes name from Nuclear Development Corp of America to United Nuclear Corp, adds reporting requirement and procedures re changes in core reactivity when reactor is shut down, "Notice of Issuance of Facility License Amendment" to UNC, dtd. 7-13-61, and Two copies of Interim Indemnity Agreement for R-49 for sign, date and return of one copy.
July 13, 61	Ltr. 7-13-61 to Oliver Townsend of Office of Atomic Development, Albany, trans. of Amend. 2 to R-49 and Notice.
July 13, 61	Ltr. 7-13-61 to Chairman of Cty. Bd. of Supervisors of Poughkeepsie, N.Y. trans cy of Lic. R-49, Amendments 1 & 2 thereto and Notice of 7-13-61.

U.S. ATOMIC ENERGY COMMISSION

APPLICANT: United Nuclear Corporation
5 New Street, White Plains, N.Y.

DOCKET NO. 50-[illegible]

FOR: Low Power (5 watt) heavy water-moderated and -reflected reactor ([illegible] Research Reactor) & Pawling Lattice Test Rig (PLATR).

SITE: NEA's Critical Facility Bldg., located at Pawling, N.Y.

DOCUMENTS

DATE DOCKETED	DESCRIPTION
July 13, 61	Ltr. 7-13-61 to ACRS trans. 3 cys. each of 7-13 Notice & Amend. 2 to R-19.
July 13, 61	Ltr. 7-13-61 to FR trans. UNC's Notice for publication.
July 13, 61	Ltr. 7-13-61 to JCAE trans. cy of Amend. 2 to R-19 & FR Notice.
July 14, 61	Ltr. 7-14-61 to United Nuclear Corp trans cy of I Agreement of 11-28-[illegible] per 7-req.
July 20, 61	UNC's Notice was filed 7-18-61, publ. 7-19-61, 26 FR 6455.
July 24, 61	Interim Indemnity Agreement of 7-13-61 returned by UNC as accepted 7-[illegible]-61 by John P. Menke.
[illegible]y 25, 61	Ltr. 7-24-61 from Marsh & McLennan trans: Endorsement 15 to NF-3, eff. 6-1-61, which amends address of insured to United Nuclear Corporation, P. O. BOX 1883, 365 Winchester Avenue, New Haven 4, Connecticut. (2 cys. ltr. & 8 cys. of end. rec'd).
	→ see page 8 for 9-27-61 ltr
Oct. 2, 61	Ltr. 9-28-61 from Marsh & McLennan trans: End. 16 to NF-3--"Amendatory Endorsement" issued 8-14-61. (1 cy ltr. & 8 cys. of end. rec'd).
Oct. 2, 61	Ltr. 10-2-61 to M&M ack. End. 16.
Oct. 24, 61	Ltr. 10-23-61 from M&M trans: End. 17 to NF-3--Earned Premium Endorsement CY 1960, issued 9-[illegible]-61. (1 cy ltr. & 8 cys. of end. rec'd).
Oct. 24, 61	Ltr. 10-23-61 from M&M trans: End. 18 to NF-3--Increase of Limit of Liability Endorsement, issued 10-[illegible]-61, increasing limit to $2,000,000 as of 10-11-61. (1 cy ltr. & 8 cys. of End. rec'd).
[illegible]. 27, 61	Ltr. 10-27-61 to M&M ack. Ends. 17 and 18 to NF-3.

(cont'd)

U.S. ATOMIC ENERGY COMMISSION

APPLICANT United Nuclear Corp.
5 New Street, White Plains, N. Y.

DOCKET NO. 50-101

[illegible]: [illegible] power (5 watt) heavy water-moderated and reflected reactor [illegible] Research Reactor) & Pawling Lattice Test Rig (PLATR)
SITE: [illegible]A's Critical Facility Bldg., located at Pawling, N. Y.

DOCUMENTS

DATE DOCKETED	DESCRIPTION
Dec. 13, 61	Ltr. 9-27-61 from UNC requesting to be advised on the status of their May [illegible], filing. (rec'd in L&R 9-29-61--rec'd in Docketing Sec 12-13-61) (Suppl. Only)
Dec. 13, 61	Ltr. 12-13-61 to UNC req them to submit within 15 days a written explanation [illegible] why deviations from NIA 3002-1 and NIA 2109-3 as listed in NIA 2131-3? [illegible] in effect during the 8-28-61, inspection were in violation of Lic R-49; and re[illegible] them to provide a desc'n of vault and shelf arrangement, etc. before complete evaluation of their appl 5-23-61.
Dec. 27, 61	Ltr. 6-26-61 from UNC replying to our ltr of 5-23-61 (to their [illegible] Development Corp of America) re Part 140. (rec'd in L&R 6-28-61--rec'd for [illegible] ing 12-27-61) (sent E3 per Huard)
Dec. 28, 61	Ltr. dtd 12/28/61 to United Nuclear Corp. requesting copies of Endorsements [illegible] thru 13, inclusive, to NELIA Policy No. NF-23 and copies of [illegible] advance premium endorsements for 21x calendar Year 1962, also requesting [illegible] to sign and return the Indemnity Agreement No. B-16 Enclosure: Indemnity Agreement No. B-16
Dec. 27, 61	Ltr. dtd 12/27/61 fm Marsh & McLennan Trans: 1. End. 20 (Advance Premium Endorsement Calendar Year 1962) to NF-[illegible] eff. 1-1-62, issued 12/1/61. 2. End. 19 (Changes in Subscribing Companies and in their proportionate liability endorsement) to NF-3, eff. 1-1-62, issued 12/1/61. 2x cy of ltr and 8 each received (not notarized)
Dec. 29, 61	Ltr. dtd 12/29/61 fm United 2x Nuclear Corp. ack our ltr of 12/13/61 and furnishing additional info as requested by us in above letter including sketch dtd 12/29/61 (Outer Buffer Fuel Element Attachment) [illegible]
Jan. 10, 62	Ltr. 1/10/62 to ACRS trans; Three copies of a letter from United Nuclear Corporation to USAEC dtd 12/29/61 regarding Lic. No. R-49
~~Jan. 18, 62~~	~~Ltr. dtd 1/12/62 to B & M requesting them to act on indemnity agreement [illegible] indemnified activities under AEC lic. Nos. [illegible], [illegible]-10, [illegible]-12 and R-4?~~ ~~which we requested in our letter of 12/6/61~~ (see 50-13)
~~Jan. 18, 62~~	~~Ltr. dtd 1/17/62 fm B & M ack our ltr of 12/29/61 to Dr. L. [illegible] [illegible] [illegible]~~ ~~[illegible] to our office. (see 50-1~~3)

U.S. ATOMIC ENERGY COMMISSION

APPLICANT: United Nuclear Corp., 5 New Street, White Plains, N. Y. DOCKET NO. 50-[illegible]

FOR: Low power (5 watt) heavy water-moderated and reflected reactor (Pawling Research Reactor) & Pawling Lattice Test Rig (PLATR)

SITE: NDA's Critical Facility Bldg., located at Pawling, N. Y.

DOCUMENTS

DATE DOCKETED	DESCRIPTION
Jan 24, 62	Ltr. 1-18-62 from Marsh & McLennan trans: (on behalf of United Nuclear Corp) 1--End No. 9 to Policy NF-3, issued 11-2-59, eff 1-1-60, "Changes in subscribing companies....." 2--End No. 10 to Policy NF-3, issued 11-2-59, eff 1-1-60, "Advance Premium [illegible] 3--End. No. 11 to Policy NF-3, issued 8-1-60, eff 12-31-59, "Earned Premium Endorsement" 4--End. No. 12 to Policy NF-3, issued 11-1-60, eff 1-1-61, "Changes in subscribing companies..." 5--End. No. 13 to Policy NF-3, issued 11-21-60, eff 1-1-61, "Advance Premium End" (1 cy. of ltr & 8 of encl rec'd)#1147
Jan. 29,62	Ltr. dtd 1/26/62 to UNC thanking them for their ltr of 12/29/61, informing us they have corrected those deficiencies in their ~~x~~ lic. program which [illegible] to their attn. in our ltr of 12/13/61 also advising them that changes to the facility beyond the scope permitted by the current license should be submitted for Commission evaluation and authorization prior to initiation. encl: ltr to Kirk fm Murphy dtd 12/29/61
Jan. 30,62	Ltr. dtd 1/29/62 to UNC trans. Amend. No. 3 to Facility License No. R-49 which authorizes UNC to possess and operate the nuclear reactor, designated as the "Pawling Research Reactor" and located at Pawling, N.Y. also advising that the auth. were among those requested in their appl. for amendment dtd 1/[illegible]/61 and the remaining auth. requested by them are now under consideration and we will advise them when action is taken~~xxxx~~ regarding them. Encls: 1. Amendment No. 3 to Lic. No. R-49 2. Notice to Office of FR 3. Hazards Analysis
Jan. 30,62	Ltr. dtd 1/29/62 to Chairman Board of Supervisors , trans. Amend. No. 3 to Fac. Lic. No. R-49 which authorizes UNC to possess and operate the nuclear reactor, designated as the "Pawling Research Reactor" and located at Pawling, N.Y. also advising that these auth. were among those re. in their appl. for amendment dtd 1/29/61 and the remaining auth. req. by them are now under consideration and will advise them when action is taken regarding them. encls: 1. Amendment No. 3 to Lic. No. R-49 2. Notice to Office of FR 3. Hazards Analysis
Jan. 30,62	Ltr. dtd 1/29/62 to Townsend trans. Amend. No. 3 to Fac. Lic. No. R-49 which authorizes UNC to possess and operate the nuclear reactor designated as Pawling Research Reactor" located at Pawling, N.Y. also advising that the auth. were among those re. in their appl. for amendment dtd 1/29/61 and the remaining auth. req. by them are now under consideration and will advise them when action is taken regarding them.

U. S. ATOMIC ENERGY COMMISSION

APPLICANT United Nuclear Corp. [illegible] **DOCKET NO.** [illegible]

FOR: Low Power (5 watt) heavy water-moderated and reflected reactor (Pawling Research Reactor) & Pawling Lattice Test Rig (Platr)
SITE: [illegible] Facility Bldg, located at Pawling, N.Y.

DOCUMENTS

DATE DOCKETED	DESCRIPTION
Jan. 30, 1962	continued from following page: encls: 1. Amendment No. 3 to Facility License No. R-49 2. Notice to Office of FR 3. Hazards Analysis
Jan. 30, 1962	Ltr. dtd 1/29/62 to ACRS trans. following documents entitled "Issuance of Amendment No. 3 to Facility Lic. No. R-49 1. Notice to Office of FR 2. Amendment No. 3 to Facility License No. ~~R-49~~ R-49 3. Hazards Analysis
Jan 31, 62	Ltr. 1-31-62 to HLW (Terry) trans cy of Amendment No 3 to Lic R-49; Notice, and Hazards Analysis.
Feb. 5, 62	Notice of of issuance of Amendment No.3, issued 1-29-62 file in FR 2-2-62; published 2-3-62; Citation 27 FR 1037....no action required (30-day intervention)
Feb. 12/62	Ltr. dtd 2/12/62 fm UNC re: our ~~xx~~ ltr of 12/28/61 enclosing two ~~xxx~~ signed copies of a formal indemnity agreement which indemnified activities under AEC Lic. No. R-49, also advising them we would appreciate their early action on this matter.
Feb. 19, 62	Ltr. dtd 2/16/62 fm Marsh & McLennan , regarding their ltr of 1/19, at which time ~~xxxx~~ they transmitted certain materia an info to us with regard to endorsements issued under the captioned policy(Nelia Policy No. NF-3) UNC [illegible] New York
Mar 22, 62	Ltr. 3-21-62 fm Marsh & McLennan advising that they assume that matl presented in their ltr of 2-16-62 is satisfactory since they haven't heard from us. (suppl only)#3130
Mar 27, 62	Ltr. 3-27-62 to Marsh & McLennan advising them that ltrs of 2-16, 3-21 and 1-18 was the proof of financial protection required re UN, but our further comm. w/UN was re. formal indemnity agreement which will we contact them re.
March 27, 1962	Telegram 3-27-62 to UNC req. them to advis us of what actio they are taking on formal indemnity agreement sent w/our lt of 12-28-61 which has never been returned by them.
Apr 11, 62	Ltr. 4-9-62 from UNC trans. 1 cy of Indemnity Agreement No. B-16 dtd 12-28-61, for act licensed under Lic R-49, signed and accepted by John R Menke, Vice President.

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WLR-5 (3-62) U. S. ATOMIC ENERGY COMMISSION

APPLICANT United Nuclear Corp. & Sabre-Pinon Corp (eff 9-13-62) 5 New Street, White Plaines, N. Y. DOCKET NO. [illegible]

FOR: Low power (5 watt) heavy water-moderated and reflected reactor (Pawling Research Reactor) & Pawling Lattice Test Rig (PLATR)

SITE: NDA's Critical Facility Bldg. located at Pawling, N. Y.

DOCUMENTS

DATE DOCKETED	DESCRIPTION
Apr 23, 62	Ltr. 4-23-62 to UNC trans: Amendment No 1 to Indemnity Agreement No. B-16 (2 cys), which implements the provisions of Public Law 87-206 and is eff as of 9-[illegible]-61, for signature and return of one cy.
May 4, 62	Ltr. 5-1-62 from UNC advising that on 4-2-62 UNC merged into and transfer all of its assets and business to Sabre-Pinon Corporation; therefore, the req that Lic R-49, amendment # 3, be further amended to transfer the lic the new corporation. Included: "Proxy Statement" dtd 2-27-62. (6 cys of each rec'd--notarized 5-1-62)#4510 (16 addl cys rec'd 5-17)
May 7, 62	Ltr. 5-3-62 from UNC returning Amendment No 1 to IA No B-16, accepted [illegible]-62 by John R. Menke. #4534
May 9, 62	Ltr. 5-9-62 to UNC req them to furnish sixteen addtl cys of their appl [illegible]-62 and Proxy Statement dtd 2-27-62.
May 17, 1962	Ltr. 5-14-62 from United Nuclear Corp trans. 16 addl. cys of 5-1-62 fil[illegible] (1 cy ltr rec'd--#4831)
May 22, 62	Ltr. 5-22-62 to Terry of HEW trans. cy trans. cy of UNC's 5-1-62 filing.
May 22, 62	Ltr. 5-22-62 to Townsend of Off of Atomic Development trans cy of UNC's 5-1-62 filing.
Aug 8, 62	Ltr. 8-6-62 from Marsh & McLennan trans the following which accommodates a reduction in the Limit of Insurance afforded to $1,00,000 for UNC: End No 21 to NF-3, "Reduction of Limit of Liability End," issued 7-[illegible]-62, eff 7-6-62. (1 cy of transl ltr & 8 cys of encl rec'd--#[illegible])
Aug 16, 62	Ltr. 8-16-62 to UNC advising them of recent amendments to Part [illegible] and [illegible]gesting that they submit proposed tech. specs for our approval if they not already have such designated in hazards summary report, in order for them to take advantage of the flexibility afforded by the amends. (form ltr)
Sept 13, 62	Ltr. 9-13-62 to UNC trans: Amendment No 4 to Lic R-49, dtd 9-13-62, which authorizes a new corporate entity also to be known as UNC which was formed by the merger of UNC and the Sabre-Pinon Corporation on 4-2-62 and in accordance with their req of 5-1-62. Related Notice, dtd 9-13-62.

WLR-5 (3-62) U. S. ATOMIC ENERGY COMMISSION

APPLICANT United Nuclear Corp. & Sabre-Finon (eff 9-13-62) 5 New Street, White Plaines, N. Y. DOCKET NO. 50-101

FOR: Low Power (5 watt) heavy water-moderated and reflected reactor (Pawling Reactor) & Pawling Lattice Test Rig (PLATR)
SITE: NDA's Critical Facility Bldg. located at Pawling, N. Y.

DOCUMENTS

DATE DOCKETED	DESCRIPTION
Sept 13, 62	Ltrs 9-13-62 to the following trans cy of Notice, dtd 9-13-62: County Board of Supervisors SG Terry of HEW Townsend (Office of Atomic Development, NY)
Sept 17, 62	Ltr 9-17-62 to OFR trans Notice of Issuance of Amendment for publicat.
Sept 19, 62	Ltr 9-19-62 to UNC furnishing guide prepared by DL&R to provide guidance on the organization and contents of HER and to describe the use of these reports by DL&R in evaluating appls for licenses--which is for their comments and suggestions (and is to be rec'd by 1-1-63) for modification to be included in a revised publication. (form ltr)
Sept 26, 62	Notice of Issuance of Amendment No 4 filed 9-20-62; published 9-21-62; Citation 27 FR 9420 (Doc. 62-9404); no action required.
Oct 25, 62	Ltr 10-18-62 from UNC to NYCO entitled, "Monthly Progress Ltr No. 12 for the period August 1, 1962 to Sept 30, 1962" (1 cy rec'd--#9994) (suppl [illegible]
Nov. 29, 62	Ltr. 11-28-62 fm Marsh & McLennan trans: End No 24 to NF-3, "Standard Prem and Reserve Prem End (Cal Yr 1961 issued 9-15-62; eff 4-16-57. (1 cy of ltr & 8 cy of encl rec'd--#10
Jan 7, 63	Ltr. 1-3-63 fm Marsh & McLennan trans: End. No. 22 to Policy No. NF-3, "Industry Credit Rating Plan Premi End," eff 4-16-57; issued 9-15-62 End. No. 23 to Policy No. NF-3, "Amendment of the Earned Premium End (Cal Years 57 thru 60)," eff 4-16-57; issued 9-15-62. (1 cy of ltr & 8 cys each encl.--#72)
Jan 7, 63	Ltr. 1-4-63 from Marsh & McLennan trans: Endorsement 27 to Policy NF-3--"Amendment of Description of Facility eff. 7-6-62, issued 12-17-62. (2 cys. ltr. & 8 cys. End. rec'd)
Jan. 14, 63	Ltr. 1-14-63 to UNC requesting copies of all endorsements, if any, to their current policy that they have not already submitted in order to aid us in keeping current on the proof of financial protection require by their license. (form ltr)

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WLR-5 (3-62) U. S. ATOMIC ENERGY COMMISSION

APPLICANT 5 New Street, White Plains, N. Y. DOCKET NO.

FCR: Low Power (5 watt) heavy water-moderated and reflected reactor (Pawling Nuclear Reactor) & Pawling Lattice Test Rig (PLATR)
LOC: [illegible] located at Pawling, N. Y.

DOCUMENTS

DATE DOCKETED	DESCRIPTION
Jan. 22, 63	Ltr. 1-21-63 fm Marsh & McLennan trans the following in conn. w. [illegible]: End No. 25 to Policy NF-3 "Changes in Subscribing Companies..." eff 1-1-63; and End No. 26 to Policy NF-3 "Advance Prem End Cal Year 1963," eff 1-1-63 (1 cy of ltr & 8 cys of each encl. rec'd--#5367) Ltr. 9-11-63 to licensee advis. of 45-day prior submtl. for NCRS review items.
July 31, 63	Ltr. 7-29-63 fm UNC advising that if there is occasion to contact Hamlin on overall corporate licensing matters, his new address is UNC, P. O. Box 4F, Centreville, Md. 21617...Also advising that this does not change the current addresses for any of UNC's divisional activities (licensing). #5367 (Also see 50-207, 70-13, -34, -371)
Oct. 28, 63	Ltr. 10-23-63 fm Marsh & McLennan trans: End No. 28 to Policy NF-3 "Address of NELIA End," eff 12-1-63; End No. 29 to Policy NF-3 "Stand Prem & Res Prem End Cal Yr 1963," eff 12-31-63. (1 cy of ltr & 8 cys of end. rec'd--#7197)
Dec. 31, 63	Ltr. 12-31-63 to UNC requesting that they submit proof of financial protection required by license for Cal Year 1964 as soon as possible. (form ltr)
Jan 29, 64	Ltrs. 1-27-64 fm Marsh & McLennan trans: End No. 30 to Policy NF-3--Changes in Subscribing Companies, eff 1-1 End No. 31 to Policy NF-3--Adv Prem and Cal Yr/1964, eff 1-1-64. (1 cy of ltrs & 8 cys of ea end rec'd--#453, 454)
Feb. 11, 64	Ltr. 2-11-64 to UNC (form ltr) advising them of AEC-Army standards for [illegible] operators... and inviting their comments on content or [illegible]

cont'd

WLR-5 (3-62) U. S. ATOMIC ENERGY COMMISSION

APPLICANT United Nuclear Corp. & Cabre-Pinon
5 New Street, White Plaines, N. Y.

DOCKET NO. [illegible]

DOCUMENTS

DATE DOCKETED	DESCRIPTION
June 19, 64	Ltr. 6-19-64 to UNC trans the following which is a result of a 6-17-64 FR ; Amendment No. 2 to Indemnity Agreement No. B-16....for acceptance. Rule Change (as published in FR) (form ltr)
July 7, 64	Amendment No. 2 to Indemnity Agreement No. B-16 as accepted by Walto S. Hamilton (?). #3610
Aug. 7, 64	Ltr. 8-5-64 fm Marsh & McLennan trans: End No. 32 to Policy NF-3--"Stand. Prem & Res Prem End f/Cal. Yr 1963," eff 12-31-63. (8 cys of end rec'd--[illegible])
Nov. 5, 64	Ltr. 11-4-64 fm Marsh & McLennan trans: End No. 33 to NF-3--"Amendment of Description of the Facility," eff [illegible] End No. 34 to NF-3--"Increase of Limit of Liability End," eff [illegible] End No. 35 to NF-3--"Amendatory End," eff 8-19-64. (8 cys of each end rec'd #5157)
Jan. 13, 65	Ltr. 1-13-65 to United Nuclear req. most recent ends. to NF-3 in order that records on proof of financial protection are kept current.
Feb. 3, 65	Ltr. 2-2-65 from United Nuclear advising that End. 35 is last end. to NF-3; 36 and 37 are in process and will be sent to us when available. (1 cy)#[illegible]
Feb. 4, 65	Ltr. 2-2-65 fm Marsh & McLennan trans: End. No. 36 to NF-3--Changes in Subscribing Companies End. No. 37 to NF-3--Advance Premium Endorsement for CY 1965. (1 cy ltr. & 8 cys each end. rec'd)#424
Sept 17, 65	Ltr. 9-15-65 fm Marsh & McLennan trans: End No. 38 to NF-3 --Stand Prem & Res Prem End f/Cal Yr 1964, eff 12-31-64. #3079
Dec. 23, 65	Ltr. 12-23-65 to UNC trans the following for their review and acceptance which has been issued as a result of amendment to 10 CFR 140 published in the FR on 11-30-65, effective 1-1-66: Amendment No. 3 to Indemnity Agreement No. B-16 (form ltr)

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APPLICANT: United Nuclear Corporation

DOCKET NO. 50-[illegible]

DOCUMENTS

DATE DOCKETED	DESCRIPTION
Dec. 29, 1965	Ltr. 12-28-65 fm UNC returning Amend No. 3 to Indemnity Agreement No. B-16 as accepted 12-28-65 by Walter A. Hamilton, Vice Pres - Admin., United Nuclear Corp. #4091
Jan. 12, 1966	Ltr. 1-10-66 fm Marsh & McLennan trans: End No. 39 to NF-3 --Changes in Sub. Companies, eff 1-1-66 End No. 40 to NF-3 --Adv Prem End f/Cal Yr 1966, eff 1-1-66 End No. 41 to NF-3 --Amend of Cond 4 End, eff 1-1-66. (8 cys of ea encl --#125) (responding to ltr of 1-7-66 --cy in file)
Jan. 11, 1966	Ltr. 1-11-66 to UNC requesting early action on the indemnity fee in connection with Lic R-49 for which they were billed 9-15-65.
Feb. 10, 1966	Ltr. 2-8-66 fm UNC (to Controller) forwarding check in payment of indemnity fee covering Lic R-49 for the one year period Oct 22, 1965 through Oct. 21, 1966. (cc to Huard)
March 28, 1966	Ltr. 3-24-66 (NDEO-641) fm UNC advising that they plan to modify their critical facility at Pawling, N. Y. to permit measurements of light water moderated fuel latticies, etc.....and trans. the following for our informal review: UNC-NDEO-624 - "Amendment II to Pawling Lattice Test Rig (PLATR) Hazards Summary Report," dated 2-28-66..... (1 cy. ltr. & 2 encl.----#900---SUPPL. ONLY)
	.....
May 13, 1966	Ltr. 5-11-66 fm UNC submitting the following for approval of Amend No. 2 to Lic R-49: UNC-NDEO-624 --(the same as above) relating to measurements of light water moderated fuel latticen. (1 cy ltr & 2 cys of rpt rec'd --not notarized) #1[illegible]
May 2, 1966	Ltr. 4-28-66 fm UNC requesting authorization to withdraw sam for use in the Pawling, NY critical facility.

(cont'd)

AEC-5 (3-65) U. S. ATOMIC ENERGY COMMISSION PAGE [illegible]

APPLICANT United Nuclear Corporation DOCKET NO. [illegible]

DOCUMENTS

DATE DOCKETED	DESCRIPTION
May 18, 1966	Ltr. 5-12-66 fm UNC superseding and amending their ltr of 5-11-66. (#1431 --not notarized)
June 3, 1966	Ltr. 5-31-66 fm UNC (NDEO-687) superseding and amending their 5-11 and 5-12-66 ltrs re planned modifications to their Pa ling critical facility, and trans the following for our approval: Report NDEO-624A (Revised) --"Amendment 2 to PLATO Hazards Summary Report," dtd 5-20-66 which constitutes Amend No. 5 to Lic R-49. (3 of ltr & 18 cys of encl rec'd notarized 6-1-66) #1604
June 8, 1966	Ltrs. 6-8-66 to Townsend and the SG trans cys of UNC's 5-31-66 filing.
July 6, 1966	Ltr. 7-6-66 to UNC trans SNM Draft authorizing allocation of 81.168 kgs of contained U-235 ....under Lic R-49, as requested 4-29-66.
July 19, 1966	Ltr. 7-18-66 fm UNC ack our 7-6-66 ltr and stating that, in view of their tight schedule, they would appreciate anything we can do to make this material available to them at the earliest possible date. (1 cy --#2103) (Mr. Doulos advised Oak Ridge by phone 7-[illegible])
August 8, 1966	Ltr. 8-4-66 fm Marsh & McLennan trans: Endorsement No. 42 to NELIA Policy No. NF-3. (1 cy. ltr & 6 cys. encl. rec'd.....#2291)
Aug. 11, 1966	Ltr. 8-11-66 to United Nuclear trans list of addtl information required in connection with the review of NDEO-624A(Revised), Amendment 2.

(cont'd)

APPLICANT United Nuclear Corporation DOCKET NO. 50-101

DOCUMENTS

DATE DOCKETED	DESCRIPTION
Sept. 26, 1966	Ltr. 9-22-66 fm UNC trans the following as a reply to our 8-11-66 ltr. in support of their 5-31-66 appl: Report --Supplement A (NDEO-770) to Amendment 2 to PLATR Haza Summary Report, NDEO-624A, dtd 9-19-66, which contai Appendix A --Technical Specifications for the UNC PLATR Appendix B --Answers to Questions (requested in 8-11 ltr) Dwgs: 16623 --Test Section Assembly/Type F 16624 --Test Section Assembly/Type V 16898 --Moderator Displacement Grid (1 of ltr & 18 cys of each encl rec'd -- 18 notarized of Suppl w/Seal & Signature) #2734
Sept. 28, 1966	Ltr. 9-28-66 to Townsend and SC/PHS (HEW, Rockville) trans cy. of UNC's 9-22-66 filing.
Dec. 12, 1966	Ltr. 12-8-66 fm Marsh & McLennan trans: End No. 43 to NF-3 -- Changes In Sub. Companies, eff. 1-1-67. End No. 44 to NF-3 -- Advance Premium Endorsement f/Cal Yr 1' eff 1-1-67. (1 cy ltr. & 8 cys. ea en rec'd. -- #3510)
Dec. 15, 1966	Ltr. 12-12-66 fm UNC requesting temporary license to permit receipt and storage of fuel rods containing up to 83.5 kg of U-235 in connection with light water lattice experiments to be performed in thei Critical Facility at Pawling. #3529 (INFO ONLY)
Jan. 27, 1967	Ltr. 1-27-67 to UNC trans: Amendment No. 5 to License R-49, dtd 1-27-67, which authoriz UNC (1) to convert its exting reactor to a light water moderated reactor so that measurements can be made of variou lattice parameters for slightly enriched oxide fuel rods in light water environment, (2) to receive, possess and use up 5500 kilograms of uranium containing up to 150 kgs of U-235 in connection with operation of the reactor and (3) incorpor tech specs in the license. Related Notice, dtd 1-27-67. Safety Evaluation, dtd 1-27

APPLICANT United Nuclear Corporation DOCKET NO. 50-101

DOCUMENTS

DATE DOCKETED	DESCRIPTION
Jan. 16, 1967	Ltr. 1-16-67 to UNC trans: Amendment No. 4 to Indemnity Agreement No. B-16, which indemnifies the actions authorized by License SNM-993 for their review, and acceptance. License SNM-993.
Jan. 25, 1967	Amendment No. 4 to Indemnity Agreement No. B-16 as accepted by William F. Roche. #267
Jan. 27, 1967	Ltrs. 1-27-67 to the following trans cys of Amendment No. 5, Notice, and Safety Evaluation: Chairman, County Board of Supervisors SO/PHS (HEW, Rockville) Townsend, OASD
Jan. 30, 1967	Ltr. 1-30-67 to OFR trans Notice of Issuance of Facility License Amendment for publication.
Feb. 10, 1967	FR Notice of Issuance of Facility License Amendment filed w/FR 2-2-67; published 2-3-67; Citation 32 F.R. 2390; AD: None (15 day post intervention)
Feb. 27, 1967	Ltr. 2-23-67 fm March & McLennan Inc. trans the following: End No. 45 to NF-3 —Standard Premium and Reserve Premium Endorsement f/Cal Yr 1966, eff 12-31-66 (1 cy ltr. & 8 cys. encl. rec'd. --#139

(cont'd.)

BETH-5 (3-65) U. S. ATOMIC ENERGY COMMISSION

APPLICANT United Nuclear Corporation DOCKET NO. 50-101

DOCUMENTS

DATE DOCKETED	DESCRIPTION
July 21, 1967	Ltr, 7-20-67, fm Marsh & McLennan trans the following: End. #47 to NF-3---Reserve prem. refund end. f/cal yr 1967 effective 7-1-67(1 cy ltr, 8 cy encls rec'd #2416
August 4, 1967	Ltr, dtd 7-31-67, fm Marsh & McLennan trans: Endorsement No. 46- Amendment of 1967 Advance Premium End. under NELIA Pol. No. NF-3 effective 1-1-67 (8 cys of endorsement rec'd.....#2576)
August 25, 1967	Ltr, 8-21-67, fm Marsh & McLennan trans the following: Endorsement No. 48 of NELIA Pol. NF-3- Amendment of Address Endorsement, eff. 7-17-67. (1 cy rec'd.....#2826)
Nov. 20, 1967	Ltr, 11-20-67, to UNC trans SNM Lic. and cy of Amendment No. 5 to Indemnity Agreement No. B-16 to be signed & returned.
Dec. 4, 1967	Ltr, 11-30-67, fm UNC returning signed and accepted cy of Amendment No. 5 to Indemnity Agreement No. B-16 as requested in ltr of 11-20-67. (1 cy rec'd.....#3914)
Dec. 12, 1967	Ltr, 12-7-67, fm UNC trans signed and accepted cy of Amendment No.6 to Indemnity Agreement No. B-16 as req in ltr of 11-[illegible]-67. (1 cy rec'd.....#4024)

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BETH-5 (3-65) U. S. ATOMIC ENERGY COMMISSION

APPLICANT United Nuclear Corporation DOCKET NO. 50-101

DATE DOCKETED	DESCRIPTION OF DOCUMENTS
Jan 24, 1968	Ltr 1-22-68 fm Marsh & MCLennan trans the following: End #49 to NF-3-Changes in subscribing Co's eff 1-1-68 End #50 to NF-3-Adv & Std Prem Notice eff 1-1-68 End #51 to NF-3 Adt of Description of facility End #52 to NF-3 Std Prem & Reserve Prem (1 cy ltr & 8 cys encls rec'd.....#244)
April 15, 1968	Ltr, dtd 4-10-68 fm UN pursuant to the req. of Lic R-49 reporting the observation of an overall moderator temperature coefficient at variance w/the req. of the tech spec..... (1 orig ltr rec'd.....#1330)
June 10,1968	Ltr, dtd 6-5-68, fm Marsh & McLennan trans the following: End No. 53 to NELIA Policy NF-3 (1 cy ltr rec'd; 8 cys end rec'd.....#2117)
June 13, 1968	Ltr, dtd 6-12-68 fm United Nuclear in ref to Lic R-49 and trans the following: AMENDMENT NO. 6 to LIC R-49 of the Pawling Lattice Test Rig and Amend # to PLATR SAR Operation w/PuO_2 UO_2 Fuel UNC-5216 (1 ltr rec'd, 3 Not. Amend 17 conf'd cys rec'd.....#2148)
June 14,1968	Ltr, dtd 6-12-68 fm United Nuclear submitted as ANNUAL REPORT .....& describing changes in the Pawling Lattice Test Rig and trans: Figure 1, Revised Poison Water System (1 orig ltr, 1 encl rec'd.....#2159
July 11, 1968	Ltr dtd 7-8-68 fm Marsh & McLennan, Inc. trans: Ltr fm March & McLennan to United Nuclear encl check rep the return allocation of the reserve premium for policy calendar year 1958 (1 cy rec'd.....#2486)

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BETH-5 (3-65) U. S. ATOMIC ENERGY COMMISSION

APPLICANT United Nuclear Corporation DOCKET NO. 50-101

DATE DOCKETED	DESCRIPTION OF DOCUMENTS
Sept 11,1968	Ltr, dtd 9-11-68, to United Nuclear Corporation trans the follow: AMENDMENT NO. 6 TO FACILITY LIC R-49 which authorizes the licensee to receive, possess, and use in the Pauling Lattice Test Rig up to 7 kilograms of plutonium in a mixture of plutonium dioxide-uranium dioxide in accordance with your application for amendment submitted 6-12-69
Sept 16,1968	Ltr, dtd 9-16-68, to OTR trans an orig and two cert cys of Notice of Issuance of Facility License Amendment
Sept 19,1968	Ltr, dtd 9-19-68 to SC/HEW trans UNC ltr of 6-12-68 req lic amend and ltr to UNC, dtd 9-11-68 granting same.
Sept 20,1968	Ltr, dtd 9-20-68, to County Bd of Supervisors of Duchess County trans Amend No. 6 to R-49, Federal Register Notice, and the Safety Evaluation.
Sept 23,1968	FEDERAL REGISTER NOTICE, dtd 9-23-68, Notice of Issuance of Facility License Amendment...Citation: 33 FR 14242.... DF: 9-19-68...DP: 9-19-68 AD: None
Sept 24,1968	Ltr, dtd 9-24-68, to New York State Dept of Commerce trans Amend #6 to R-49, Federal Register Notice, Safety Evaluation, and UNC's ltr of 6-12-68 requesting same.
Sept 30,1968	Ltr, dtd 9-30-68, to receipients of amend #6 to License enclosing corrected Attachment A
Sept 30,1968	Ltr, dtd 9-30-68, fm Compliance Region II trans: CO Report No. 50-111/68-2 reporting on the inspection of Sept 11-12, 1968.
Oct 29, 1968	Ltr, dtd 10-29-68, to United Nuclear trans Form AEC-592
Nov 14,1968	Ltr, dtd 11-8-68, fm United Nuclear trans as reply to Form AEC-592

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BETH-5 (3-65) U. S. ATOMIC ENERGY COMMISSION

APPLICANT United Nuclear Corporation DOCKET NO. 50-101

DATE DOCKETED	DESCRIPTION OF DOCUMENTS
Dec. 2, 1968	Ltr. 12-2-68 to UNC advising that during the period of Aug. 12-19, 1968, a safeguards inspection of the snm held pursuant to SNM-871, R-49 and CX-25 was conducted..... and trans: "Report on the Inspection of Safeguards Control of Nuclear Materials of UNC" (See 70-903 and 50-290 also)
March 6, 1969	Ltr. 3-3-69 fm UNC trans: Endorsements 54 thru 61 to NF-3 (1 cy. ltr. & 8 encl.----#775)
June 24, 1970	Ltr. 6/18/70 fm UNC trans. the following in reply to our 6/8 ltr. Endorsement Nos. 1 thru 7 and 62 thru 64 to Policy No. NF--3 (1 cy. rec'd----------#1853)
Jan 11, 1971	Ltr 1-4-71 fm UNC trans the following: Endorsement No. 67 to NELIA Policy No. NF-3 (1 cy ltr & 8 cys encl rec'd.....#74)
June 23, 1971	Ltr 6-23-71 to UNC adv that explosives in a reactor facility must be evaluated carefully because of the potential for damage to the reactor
Continued	

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BETH-5 (3-65) U. S. ATOMIC ENERGY COMMISSION

APPLICANT: United Nuclear Corporation DOCKET NO. 50-101

DATE DOCKETED	DESCRIPTION OF DOCUMENTS
June 30, 1971	Ltr 6-29-71 fm UNC notarized adv Gulf Oil & UNC will form a joint venture corporation commencing business as Gulf United Nuclear Fuels Corp and trans: ATCHMT "A"- Partial Organization Chart ATCHMT "B"- Resume of Arnold R. Fritsch ATCHMT "C"- Resume of Kenneth L. Wiley List of Principal Officers 1970 Annual Fin Rpt for UNC & Gilf Oil (3 signed Orig & 19 conf'd cys, 22 cys encl rec'd, and 7 cys Annual Rpt.....#2986)
July 7, 1971	Ltr 7-1-71 fm UNC notarized req trans of lic R-49(50-101, & CX-25(50-290).....be trans to GUNFC in accord with UNC ltr dtd 6-29-71 (3 signed & 19 conf'd cys.....#3040)
July 7, 1971	Ltr 7-1-71 fm GUNFC notarized 7-1-71 req trans of lic R-49 (50-101) & CX-25 (50-290) in accord with UNC ltr dtd 6-29-71... List of Directors of Gulf United Nuclear Fuel Corp. (3 signed & 19 conf'd.....#3049)
Aug. 2, 1971	Ltr 7-22-71 fm United Nuclear Corp notarized 7-22-71 re their 6-29-71 & 7-1-71 ltr....furnishing addl info in support of requests & trans the following: (1) Gulf United Nuclear Fuels Corp Pro-Forma Balance Sheet as of 3-31-71..... (2) Lease Agreement w/attchmt's Exhibit A & B..... (3 signed & 19 conf'd cys of ltr & 22 cys ea encl rec'd.....#3484)
Oct. 29, 1971 INPUT	Ltr 10-28-71 fm GUNFC notarized 10-28-71 requesting TERMINATION OF LIC. R-49...& trans the following: REPORT: Radiation Survey of PLATR (3 signed & 22 conf'd cys of ltr & 25 cys encl rec'd.....#4730)

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BETH-5 (3-63) U. S. ATOMIC ENERGY COMMISSION

APPLICANT: United Nuclear Corporation DOCKET NO. 50-151

DATE DOCKETED	DESCRIPTION OF DOCUMENTS
Dec 15, 1971	Ltr 12-15-71 to UNC adv that we would appreciate an early decision as to the disposition that wish to be made of their request of 10-28-71
Jan 17, 1972	Ltr 1-13-72 fm GUNFC re our 12-15-71 ltr....req that joint appl of Gulf United Fuels Corp & United Nuclear Corp for transfer of license R-49 & CX-25 be continued to be processed... (1 Orig.....#260)
Feb 17, 1972	Ltr 2-8-72 fm GUNFC notarized 2-8-72....request joint appl of GUNFC & UNC be treated as a Class 104 license (3 signed notarized & 19 conf'd.....#600)
Mar 13, 1972	Ltr 3-9-72 fm UNC re ply to our 10-5-71 ltr request for info on past experience with annual levels radioactivity (1 signed Orig & 10 cys encl rec'd.....#1
Apr 25, 1972 ***	Ltr 4-25-72 to UNC trans Amdt #2 to License No. CX-25; Amdt # 7 to License No. R-49; Amdt No. 11 to Indemnity Agreement No. B-16 and Federal Register Notice
Nov 9, 1972 ***	Ltr 11-6-72 fm GUNF adv Mr. Robert Kopp is responsible for all correspondence and req all communications be directed to Mr. Kopp at the address given.... (8 cys rec'd.....#6157-
Nov 20, 1972 ***	Ltr 11-6-72 fm GUN notarized 11-6-72 req an AMDT to Lic R-49 in order to change in status of PLATR..... (3 signed ltr.....43 cys rec'd.....#63
Dec 21, 1972	Ltr 12-21-72 to GUNF in response to 11-6-72 ltr and adv that fees of $1,500 & $3,500 are due and should be remitted promptly

APPLICANT: United Nuclear Corporation

DOCKET NO. 50-101

DATE DOCKETED	DESCRIPTION OF DOCUMENTS
June 12, 1973 ***	Ltr 6-8-73 fm GUNFC notarized 6-8-73 furnishing info re appl to amend facility license R-49 replacing 11-6-72 submittal (NISM-72-92) & trans Appendix A to Lic. No R-49 - Tech Specs for GUNFC Pawling Lattice Test Rig dtd 6-8-73. (3 signed orig.....40 cys encl rec'd.....#3793)
June 27, 1973 ***	Ltr 6-27-73 to UNC ack ltr dtd 6-11-73 reporting suppl info on exposure reported in ltr dtd 3-22-73
Oct 11, 1973	Memo 10-11-73 to D.J.Skovholt, AD-OR(L) fm H.D.Thornburg, C hief FSEB(L) conversion of Lic No R-49 & CX-25 fm oper to posession.
Oct 25, 1973 ***	Ltr 10-23-73 fm GUNC notarized 10-23-73 adv AEC of their intent to terminate Lic Nos R-49 & CX-25 & to dismantle & decommission both the PLATR & PTF.....w/attched tables 1 thru 3. (3 signed orig.....39 cys rec'd.....#7811
Nov 5, 1973 ***	Ltr 11-1-73 fm GUNF notarized 11-1-73, re their 10-23-73 appl, furnishing addl info to Pawling Lattice Test Rig (PLATR) & Pawling Test Facility (PT Dismantling Plan. (4 signed orig.....39 cys rec'd.....#8009)
Nov 14, 1973 ***	Ltr 11-14-73 to GUNFC re their ltr 10-9-73 in responce to our 9-21-73 ltr informing us of steps taken to correct items of noncompliance.
Nov 21, 1973 ***	Ltr 11-19-73 fm Gen Atomic Co notarized 11-19-73 req written consent to aquire by transfer fm Gulf Oil Corp all rights, title & interest, whether direct or indirect of Gulf in lic or appl for lic & trans..... (1) APP "A" lic or appl for lic which are held or have been made in name of Gulf or div of Gulf (GEES, GGAC, GUNF & AGNS) (2) APP "B" info pertinent to ident of appl & general partners. (3 signed orig.....28 cys rec'd.....#8385)

APPLICANT: UNITED NUCLEAR CORPORATION DOCKET NO. 50-101

DATE RECEIVED	DESCRIPTION OF DOCUMENTS
Dec 5, 1973 ***	Ltr 11-30-73 fm UNC re their 10-23-73 ltr & trans: Ltr dtd 10-23-73 fm Gulf UNC to D. J. Skovholt adv AEC of plans to dismantle & decomission both the PLATR & PTF reactors (1 signed.....23 cys rec'd.....#8655)
Dec 18, 1973 ***	Ltr 12-14-73 fm GAC re their appl dtd 11-19-73, with respect to the transfer fm GOC of its right, title & interest in various licenses & appls for licenses & trans the following: Resolution of the Standing Committee of the Partnership Committee adopted at a mtg thereof held on 12-14-73 (1 signed....1 cy encl.....#9006).
Dec 20, 1973 ***	Ltr 12-20-73 to GUNFC re appl dtd 10-23-73 & ltrs dtd 11/1 & 30/73 & trans Notice of Intent to Issue Order Authorizing Dismantling of Fac. & SER
Jan 2, 1974 ***	Ltr 12-26-73 fm GUNFC re our 11-30-73 ltr...adv that the Appl does not intend to submit a security plan by 1-7-74... (1 signed.....#55)
Feb 4, 1974 ***	Ltr 2-4-74 to Warren R. Walworth re ltr 12-10-73, to Pre Nixon re concern on the use of a certain "nuclear fuel testing site (DR-6578)
Mar 8, 1974 ***	Ltr 3-8-74 to UNC trans Amdt to Indemnity Agreement & Amdt to 10 CFR Part 140
Mar 29, 1974 ***	Ltr 3-13-74 fm GUNF notarized 3-13-74, requesting termination of lic nos R-49 & CX-25, trans the following: Results of Radiological Survey of Critical Facility Bldg - PLATR and TTF (1 signed [illegible]

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***DENOTES SENT TO PDR
****DENOTES SENT TO LPDR

BETH-5 (3-65) U. S. ATOMIC ENERGY COMMISSION PAGE: 27

APPLICANT: UNITED NUCLEAR CORPORATION DOCKET NO. 50-101

DATE RECEIVED	DESCRIPTION OF DOCUMENTS
Apr. 26, 1974 *** ****	Ltr to GNFC re the inspection conducted by Mr. Jerman on Apr. 9-11 and 15-16, 1974 at Pawling, New York of activities authorized by AEC Lic. Nos. SNM-871, R-49 and CX25 .
June 25, 1974 ***	Ltr 6-25-74 to GUNFC trans Lic Termination Order, Amdt #13 to Indemnity Agreement No. B-16.
July 16, 1974	Ltr 7-16-74 fm UNC trans Amdt #13 to Indemnity Agreement No B-16.
Aug 13, 1974 ***	Ltr 8-13-74 to UNC adv hav not rec'd endorsements 75, 76, 77, 81 & 82 to NF-3.
Mar 21, 1975 ***	Ltr 3-21-75 to GUNFC trans (1) Amdt to Indemnity Agreement, (2) Amdt to 10 CFR Part 140
May 2, 1975 ***	Ltr 5-2-75 to UNC trans Amdt No. 13

UNITED NUCLEAR
CORPORATION
RESEARCH AND ENGINEERING CENTER

GRASSLANDS ROAD
ELMSFORD, NEW YORK 10523
914 LYRIC 6-6000

In reply refer
to: H&S-2394

December 23, 1970

Mr. Donald A. Nussbaumer
Fuel Fabrication and Transportation Branch
Division of Materials Licensing
United States Atomic Energy Commission
4915 St. Elmo Place
Bethesda, Maryland 20014

Ref: (1) Docket No. 70-903
(2) SNM-871
(3) LIC 64-2 (Revised) dated December 5, 1966

Dear Mr. Nussbaumer:

This is a request to amend our license to permit the installation of two (2) additional chemistry glove boxes in the Plutonium Laboratory at Pawling (see Figure #3 of Reference #3). The new boxes will be attached to the existing wet chemistry box as shown in the enclosed figure (14A represents the existing wet chemistry box and 14B and 14C represents the new chemistry boxes). These three boxes will be treated as a single wet zone with a total mass limit of 250 gms. of Pu + U^{235}. This increase in the physical size of zone 22 does not alter the conclusion of the interaction analysis previously submitted.

The Wet Chemistry Glove Box II (S-401) is being installed in the new low bay area of the Pawling Plutonium Laboratory. The glove box, shown as 14B in Figure 1, is part of the new installation designed to increase analytical chemistry capacity.

The Chemistry Box (S-401) is an air atmosphere glove box constructed of stainless steel and containing six 3/8-inch thick full-view safety-glass panels for the operating sides. The interior chamber is divided into two sub chambers separated by a 3/8-inch steel bulkhead; the sub chambers have the following dimensions:

The larger "right-hand" chamber is approximately 42" wide and 6' long. The smaller chamber is approximately 42" wide and 3' long. A 16-inch diameter hole with a gas-tight seal door located in the bulkhead provides communication between the chambers.

A/70
4159

(B) 9103180153

Sixteen rings, serving as glove ports, are mounted on the glove panels to provide 8 operating stations. All interior metal surfaces are coated with liquid tile to resist chemical attack. The box is capable of being evacuated to a pressure of -8" of water below ambient. The maximum helium leak rate is 1×10^{-5} std. cc He/sec.

The overall dimensions of the box are approximately 42" wide, 41" high, and 9' long. Access into and egress out of the box are through the following ports located at the ends of the box:

East End - one 8" transfer port

West End - one 16" transfer port
- one 4" sphincter port

Bottom - one 8" pouching port

All of the ports except the 8" pouching port are closed off internally with sealed doors. The 8" pouching port is closed off with an "O" ring sealed stainless steel plate screwed to the floor of the box. The glove box is interconnected to adjacent box, S-402 (Zone 14C) and S-307 (Zone 14A) by stainless steel transfer tunnels.

All gas lines penetrating the box walls are protected by absolute filters.

Penetrating the top of the box are stainless tubes for feeding batch quantities of liquids into the box. The external ends of the tubes are attached to outlets of closed glass or plastic supply bottles. The outlets are located at the bottom of the bottles so that the end of the tube is always under a head of liquid. The liquid supply into the box is controlled by two manual valves for each tube, one being located outside of the box and the other inside of the box at the free end of the tube. The bottles are located in racks permanently mounted on the top of the box. Replenishment of the liquids will be by filling the supply bottles without removing them from the rack.

The glove box will be used for the chemical analysis of samples containing plutonium. Typical operations expected to be performed in this box consist of

1) Dissolution of samples containing plutonium and uranium in mixtures of mineral acids;

2) Dissolution of samples containing uranium and plutonium by fusion techniques;

3) Electrometric determination of uranium and plutonium;

4) Colorimetric determination for nitrogen by a modified Kjeldahl distillation method;

5) Halogen determination by pyrohydrolysis and solid state electrode detection;

6) Solvent extraction preparations for colorimetric analysis;

7) Colorimetric analysis for nickel; and

8) Solution concentration by distillation techniques.

The glove box has been designed to operate from the existing Plutonium Facility once-through air system.

The gas control system will automatically keep the pressure of the glove box at -0.5 to -1.0" H_2O during normal operating conditions with a constant supply flow of 5 to 10 cfh.

Emergency conditions which the system will handle include a glove or pouching bag rupture. Should there be a complete rupture of a glove, the exhaust system will maintain an air inflow of 100 ft/min through an 8" glove port.

The operation of the glove box on the once-through system will be identical to that of the Wet Chemistry Box S-307 in the existing facility.

The pressure in the glove box will be sensed and controlled by the existing pressure control system. The existing system contains a glove box high pressure alarm.

If a vacuum source is to be used in the glove box the pump will be controlled by a pressure sensitive switch. This switch will sense excessive vacuum, disconnect power to the pump, and require a manual restart of the pump.

All of the electrical components of the laboratory gas systems are on the automatic start generator bus system and the loss of commercial electrical power would not present a hazardous condition.

All materials transferred into the box will be through the use of the sphincter port or transfer tunnels for adjacent boxes. All

materials transferred out of the box will be by pouching through the 8-inch transfer port and the transfer tunnels.

The Balance Glove Box (S-402) is being installed in the new low bay area of the Pawling Plutonium Laboratory. The glove box, shown as (14C) in Figure 1, is part of the new installation designed to increase analytical chemistry capacity.

The Balance Box (S-402) is an air glove box made of stainless steel and two 3/8-inch thick full-view safety-glass panels for the operating sides. Four rings, serving as glove ports, are mounted on the glass panels to provide two operating stations.

The box is capable of being evacuated to a pressure of -8" of water below ambient. The maximum helium leak rate is 1×10^{-5} std. cc He/sec.

The interior dimensions of the box are 42" wide, 41" high, and 4' long. Access into and egress out of the box are through the following ports located at the ends of the box:

West End

a) one 16" pouching port - outside "O" ring sealed cover

b) one 4" sphincter port - outside sealed cover

East End

a) one 16" pouching port - couples to another glove box

b) one 4" pouching port - outside "O" ring sealed cover

All of the ports are closed off with sealed doors.

All lines penetrating the box walls are protected by absolute filters.

The glove box will be used for the preparation, weighing and storage of samples for characterization of Pu materials.

The glove box has been designed to operate from the existing Plutonium Facility once-through air system.

The gas control system will automatically keep the pressure of the glove box at -0.5 to 1.0" H_2O during normal operating conditions with a constant supply flow of 5 to 10 cfh.

Emergency conditions which the system will handle include a glove or pouching bag rupture and a sudden reduction in pressure caused by a rapid unfolding of the 16" pouching bag.

The operation of the glove box on the once-through system will be identical to that of the boxes in the existing facility.

The pressure in the glove box will be sensed and controlled by the existing pressure control system.

The existing system contains the following alarm:

a. Glove box high pressure

Should there be a complete rupture of a glove, the exhaust system will maintain an air inflow of 100 ft/min through an 8" glove port.

If a vacuum source is to be used in the glove box the pump will be controlled by a pressure sensitive switch. This switch will sense excessive vacuum, disconnect power to the pump, and require a manual restart of the pump.

All of the electrical components of the laboratory gas systems are on the automatic start generator bus system and the loss of commercial electrical power would not present a hazardous condition.

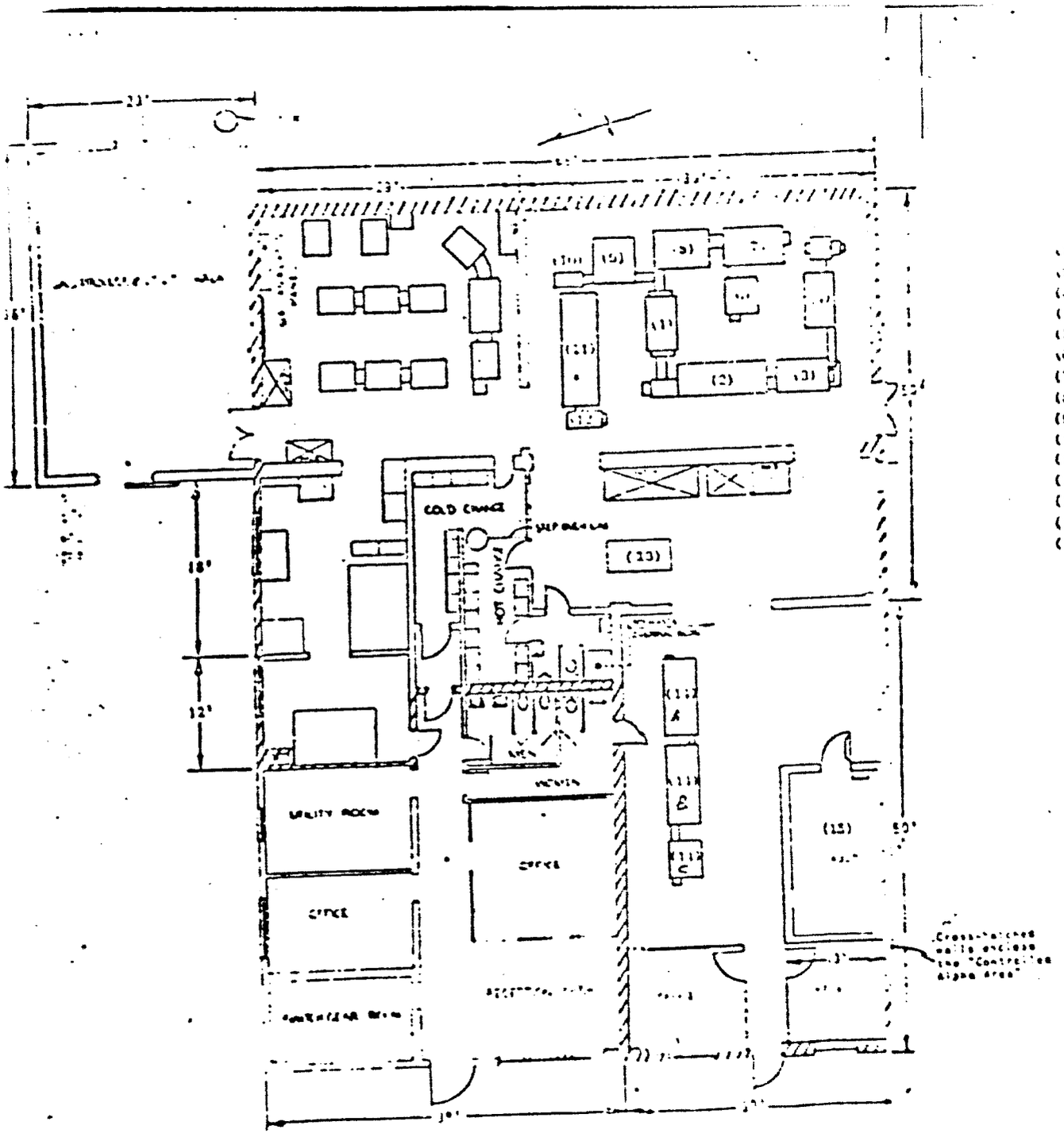
All materials transferred into the box will be through the use of the sphincter port or by standard laboratory pouching techniques. All materials transferred out of the box will be by pouching.

Your usual prompt consideration will be appreciated.

Very truly yours,

Justin Karp, Manager
Finance and Administrative
Operations

heb



Locations	Measurement Stations
(1) Blending	\
(2) Pilot fabrication	\
(3) Pilot furnace	\
(4) Grinding	-
(5) Cleaning	\
(6) Weighing	\
(7) Blend-agglomerate	-
(8) Dry-granulate	\
(9) Press	\
(10) Debond	-
(11) Sinter	-
(12) Sinter discharge	\
(13) Load and weld	-
(14) Analytical chemistry	-
(15) Storage	-

Fig. 1 — Pawling Plutonium Laboratory

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

DEC 15 1969

DML:CER
70-903
SNM-871 Amendment No. 71-6

United Nuclear Corporation
Research and Engineering Center
Wasslands Road
Wilmsford, New York 10523

Attention: Mr. Justin Karp, Manager
Finance and Administrative Operations

Gentlemen:

Enclosed is Amendment No. 71-6 to Special Nuclear Material License No. SNM-871 authorizing the delivery of special nuclear material to a carrier for transport in the PNL Model 55 package.

Please note that this amendment does not authorize the transport of special nuclear material. Such transport is normally subject to regulation by the Department of Transportation (DOT). Questions regarding their requirements should be directed to DOT.

Sincerely,

Original Signed by
Donald A. Nussbaumer

Donald A. Nussbaumer, Chief
Source & Special Nuclear
Materials Branch
Division of Materials Licensing

Enclosure:
As stated

cc: Mr. William A. Brobst
Department of Transportation

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UNITED STATES
ATOMIC ENERGY COMMISSION

LICENSE AMENDMENT
for
DELIVERY OF SPECIAL NUCLEAR MATERIAL
to a
CARRIER FOR TRANSPORT

Pursuant to the Atomic Energy Act of 1954 and Title 10, Code of Federal Regulations, Chapter 1, Part 70, and Part 71, the following amendment to the special nuclear material license identified below is hereby issued, authorizing the licensee to deliver special nuclear material to a carrier for transport, and is subject to the conditions specified in that license and to the conditions specified below.

LICENSEE	
1. Name: United Nuclear Corporation 2. Address: Research and Engineering Center Grasslands Road Elmsford, New York 10523	3. License No. SNM-871 Amendment No. 71-6 4. Docket No. 70-903

CONDITIONS

5. (a) Packaging

(1) Model number — PNL Model 55

(2) Description — Containment vessel is a norminal 4" diameter Schedule 40 stainless steel pipe, 76 1/2" long, with a bolted and gasketed "grayloc" blind flange closure on one end and a seal welded plate on the other end. Containment vessel is centered and supported within an outer 18" x 18" x 81" steel angle framework by 3/16" thick steel support plates and 1" Schedule 80 pipes. The outer framework is covered by expanded metal mesh. Container is constructed in accordance with Pacific Northwest Laboratory Drawings H-3-26549 and H-3-26550.

(b) Contents

(1) Type and form of material — Non-irradiated fuel rods containing PuO_2-UO_2 or UO_2 as ceramic pellets fully clad in leaktight stainless steel of minimum .015" thickness, zircalloy of minimum .020" thickness or other material of at least equivalent structural strength and thermal resistance. Uranium may be enriched to any degree in the U-235 isotope unless otherwise specified.

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LICENSEE: United Nuclear Corporation

PAGE NO: 2

LICENSE NO: SNM-871

AMENDMENT NO: 71-6

DOCKET NO: 70-903

(2) Maximum quantity of material per package		Total contents not to exceed 150 pounds and:
	(i)	No more than 3 w/o PuO_2 in natural or depleted UO_2, or
	(ii)	2.7 kilograms fissile material with no more than 25 w/o PuO_2 in natural or depleted UO_2, or
	(iii)	1.0 kilogram fissile material.
(c) Fissile Class		II
(1) Minimum transport index to be shown on label for Class II	(i)	For the contents described in 5(b)(1) and limited in 5(b)(2)(i) or 5(b)(2)(ii): 2.0
	(ii)	For the contents described in 5(b)(1) and limited in 5(b)(2)(iii): 2.5

REFERENCES

Licensee's application dated December 3, 1969, requesting approval to deliver special nuclear material to a carrier for transport in the PNL Model 55 package.

Battelle Memorial Institute's application dated August 21, 1969, and supplement dated September 26, 1969, Docket 70-984, requesting approval to deliver special nuclear material to a carrier for transport in the PNL Model 55.

FOR THE ATOMIC ENERGY COMMISSION

Original Signed by
Donald A. Nussbaumer

Donald A. Nussbaumer
Division of Materials Licensing

Date of Amendment DEC 15 1969

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

March 8, 1974

Donald J. Skovholt, Assistant Director for Operating Reactors, L

WITHHOLDING OF PROPRIETARY INFORMATION PURSUANT TO SECTION 2.790 FILED BY GULF UNITED NUCLEAR FUELS CORPORATION RE DOCKET NO. 50-10, ET AL

By letter dated June 28, 1973, the Gulf United Nuclear Fuels Corporation (GUNFC) transmitted proprietary Report GU-5300 - "Gulf United Light Water Reactor Fuel Rod Thermal-Mechanical Analysis Methods" (dated June 25, 1973) and requested that the report be withheld from public disclosure pursuant to Section 2.790(b) of 10 CFR Part 2. By letters dated October 15, 1973, GUNFC transmitted a proprietary version of Revision 1 of GU-5300 and proprietary answers to questions asked in Mr. Stello's letter dated August 31, 1973 and September 4, 1973, regarding review of the June 1973 filing of GU-5300, and requested that they too be withheld pursuant to Section 2.790(b). With a letter dated November 7, 1973, GUNFC filed a non-proprietary version of Revision 1 of GU-5300. Both the non-proprietary and proprietary versions of Revision 1 are revisions in their entirety of the proprietary version transmitted by GUNFC's letter dated June 28, 1973. Since these versions superseded in its entirety the original report, the original report dated June 25, 1973, is being destroyed except for the Regulatory record copies in Docket Files 50-10 and 50-29 and those copies retained by Regulatory's Technical Review personnel. By letter dated June 26, 1973, Commonwealth Edison Company (CEC) also filed proprietary information on GUNFC fuel which was approved for withholding by AEC letter dated October 2, 1973. The non-proprietary version of the June 26th filing was submitted by CEC's letter dated August 21, 1973.

In its letters dated June 28 and October 15, GUNFC stated that the documents contain certain information which could enable a reload fuel supplier in competition with them to assess current GUNFC design and manufacturing methods and calculational results and to examine the test results performed by GUNFC at its own expense, and that disclosure of this information would result in a competitive and economic disadvantage to GUNFC.

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After reviewing the proprietary and non-proprietary versions of the documents and attempting to achieve an effective balance between legitimate concerns for protection of Gulf United's competitive position and the right of the public to be fully apprised as to the operation of light water reactors using GUNFC fuels and safety issues associated therewith, we have concluded that GUNFC has made a sufficient showing that its interest could be adversely affected by disclosure of the information contained in the proprietary documents. We also have concluded that sufficient information is contained in the non-proprietary versions filed with letters dated August 21, 1973 and November 7, 1973, and in other non-proprietary documents in the public record to advise an interested member of the public on the operation of light water reactors such as Dresden Unit 1 using GUNFC fuels and applicable safety considerations associated therewith.

Based on the above, I have concluded that disclosure of the proprietary documents transmitted by GUNFC letters dated June 28 and October 15, 1973, is not required in the public interest nor by 10 CFR Part 9, and they should therefore be withheld from public inspection pursuant to 2.790(b) of 10 CFR Part 2. A letter is being sent to GUNFC granting the June 28 and October 15, 1973, withholding requests.

Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Directorate of Licensing

cc: Commonwealth Edison Company (50-10)
Yankee Atomic Electric Company (50-29)
Mr. Anthony Z. Roisman
Docket File
AEC PDR
Branch Reading
RP Reading
D. J. Skovholt
ACRS (16)
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D. Ross
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L. Rubenstein
S. Hanauer
R. Lobel

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

March 8, 1974

Docket No. 50-10

Gulf United Nuclear Fuels Corporation
ATTN: Mr. John Peak, Staff Consultant
Grasslands Road
Elmsford, New York 10523

Gentlemen:

The Commission's Regulatory staff has completed its review of your Report GU-5300 - "Gulf United Light Water Reactor Fuel Rod Thermal-Mechanical Analysis Methods" (dated June 25, 1973), Revision 1 of the report (dated October 15, 1973), and other associated items. Our evaluation is contained in the enclosed "Technical Report on Densification of Gulf United Nuclear Fuels Corporation - Fuels for Light Water Reactors", dated November 21, 1973. A copy of our evaluation and the non-proprietary Revision 1 of GU-5300 have been placed in the Commission's Public Document Room.

If Gulf United desires to discuss the results of our review, we will be happy to meet with your representatives.

Your letter dated June 28, 1973, transmitted a proprietary version of GU-5300 and your letters dated October 15, 1973, transmitted a proprietary version of Revision 1 of GU-5300 and proprietary answers to questions asked in Mr. Stello's letters dated August 31, 1973 and September 4, 1973, and requested that these items be withheld from public disclosure pursuant to Section 2.790(b) of 10 CFR Part 2. Your letter dated November 7, 1973, transmitted a non-proprietary version of Revision 1 of GU-5300.

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The reasons for the withholding requests were stated to be that the subject proprietary documents contain design and manufacturing methods, and calculational results which were developed at the expense of Gulf United and their disclosure would result in a competitive and economic disadvantage to your Corporation. The Regulatory staff has examined the proprietary documents and has determined that they qualify for withholding from public disclosure. Accordingly, pursuant to Section 2.790(b), we are withholding these proprietary documents from public inspection. Withholding of this information from public inspection shall not affect the right, if any, of persons properly and directly concerned to inspect the documents.

Sincerely,

Donald J. Skovholt
Assistant Director
for Operating Reactors
Directorate of Licensing

Enclosure:
Technical Report dtd 11/21/73

cc w/enclosure:
Commonwealth Edison Company
ATTN: Mr. J. S. Abel
Nuclear Licensing Administrator -
Boiling Water Reactors
Post Office Box 767
Chicago, Illinois 60690

Yankee Atomic Electric Company
ATTN: Mr. Donald E. Vandenburgh
Vice President
20 Turnpike Road
Westboro, Massachusetts 01581

Mr. Anthony Z. Roisman, Esquire
Berlin, Roisman and Kessler
1712 N Street, N. W.
Washington, D. C. 20036

GENERAL ATOMIC COMPANY
P.O. BOX 81608
SAN DIEGO, CALIFORNIA 92138
(714) 453-1000

NOV 6 1974
REGULATORY OPERATIONS

In Reply
Refer To: 871-484

October 9, 1974

Mr. L. C. Rouse, Chief
Fuel Fabrication & Reprocessing Branch No. 1
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: AEC License No. SNM-871; Docket 70-903

Dear Mr. Rouse:

In Mr. Peter Loysen's letter dated March 11, 1974, GAC requested that the Pawling, New York facility be deleted as an authorized place of use under the subject license. Deletion rather than termination was requested because of the subsequent transfer of authorization of the company's New Haven, Connecticut activities onto that license. Inasmuch as a request for termination of the licensed activities at New Haven has now been made, you should consider the original request as one for deletion or termination, depending upon whichever location you complete action on first.

With regard to soil sampling and analysis at Pawling, you are aware that two sets of measurements were made at different times at 15 separate locations on the site. The first set, made by General Atomic in October 1973, was used in the radiological survey report and showed no detectable Pu-238 and Pu-239. The second set, made by ATCOR in January 1974, was not available for use in the report and showed some positive but anomalous results. The anomalies appeared to be in the 239/238 ratios and more specifically, as was determined later, in the Pu-238 results themselves. With the exception of four selected sample locations and one other individual location, all of the other measurements from both sets were less than 1.0 d/m/gm Pu-239. We consider these other measurements to be reasonable and acceptable. Since no Pu-238 was ever used by itself at Pawling and no reason to suspect isotopic fractionation in soil, we must consider the Pu-238 results to be invalid.

As for the four selected sample locations, the ATCOR samples were re-analyzed by an AEC laboratory and found to be less than 1.0 d/m/gm Pu-239 with typical 239/238 ratios. Additional remeasurements of these locations by New York State Department of Environmental Conservation were found to be less than 1.0 d/m/gm also. Additional measurements of these locations by Regulatory Operations, Region I, were found to range up to 12.2 d/m/gm Pu-239. While we do not necessarily understand the reason for these higher values, we nevertheless decided to remove soil from the areas in question and to remeasure as described in Mr. Loysen's letter of August 12, 1974.

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Resampling was performed or September 5, 1974 in accordance with procedures agreed to by Regulatory Operations, Region I, who monitored the sampling. Several samples of soil were composited for each of three areas located: (a) behind the Plutonium Laboratory Building; (b) below the windows of the E-Spec room; and (c) at a low spot off the paved area southwest of the Engineering Building. Results of the analyses performed on these samples are as follows:

	d/m/gm Dry Soil	
	Pu-238	Pu-239+240
Behind Pu Lab Bldg.	<0.002±.002	0.018±.004
Below E-Spec Windows	<0.002±.002	0.002±.002
Low Spot Off Paving	<0.002±.002	0.006±.002

We believe that these results demonstrate conclusively that the actions taken to further reduce residual contamination below your guidelines were effective and that all regulatory requirements have been met. Your prompt action to terminate SNM-871 will be appreciated.

Very truly yours,

Douglas T. Farney
Licensing Administrator
General Atomic Company

DTF/lm

REPORT OF INCIDENT
AT GULF UNITED'S PLUTONIUM FACILITY
AT PAWLING, NEW YORK

January 28, 1972

GULF UNITED
NUCLEAR FUELS CORPORATION
Elmsford, New York

A/84

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1. INTRODUCTION

This report of an incident which occurred at Gulf United's Plutonium Facility located in Pawling, New York, on December 29, 1971, is being submitted in compliance with 10 CFR 20.495. The incident was reported immediately by telephone and telegram to the Director, Region 1, Division of Compliance, United States Atomic Energy Commission on December 29, 1971, in compliance with 10 CFR 20.403.

On December 29, 1971, at approximately 12:35 p.m., a pressure excursion occurred in three interconnected analytical glove boxes. The incident caused some uranium-plutonium oxide mixtures to be distributed in the portion of the laboratory where the glove boxes are located, resulting in plutonium contamination that exceeded the limits for release of radioactive materials as specified in 10 CFR 20.403. No fire resulted from the incident. No employees were present in the plutonium handling area at the time of the incident and, therefore, no individual was exposed to airborne plutonium activity or radiation. Our investigation shows that there was no release of plutonium to the environs. No equipment was damaged as a result of the incident; however, equipment contamination in this area was considerable. Decontamination is proceeding satisfactorily at this time, without further incident.

2. DESCRIPTION OF THE INCIDENT

At approximately 12:35 p.m. on December 29, 1971, a pressure excursion in three interconnected glove boxes (identified as S-202, S-203, and S-204 in Fig. 1) blew off a plastic pouch covering a 16-inch diameter port and expelled three sphincter cans with their original contents of loose UO_2-PuO_2 powder containing about 10.8 g of plutonium, about 30 empty plastic vials and friction caps, and about 8 Kimwipes onto the laboratory floor in the direction of the arrow shown on Fig. 1 of Appendix A. The airborne contamination alarm was sounded immediately by a continuous type air sampler located in the northeast corner of the Facility, called the Research Area. Two employees entered the change room immediately with face masks and observed visible evidence of the pressure excursion through the window of the main entrance door to the Research Area. The 16-inch plastic pouch cover for Box S-204, and several inner glove rings, were lying on the laboratory floor. The 16-inch plastic pouch, which had previously been rolled up and taped at the end of Glove Box S-204, was now projecting out from the box in the fully-extended, but collapsed, position. Glove Boxes S-202 and S-203 were not visible from the viewing location. No employees were in the plutonium handling area at the time of the incident. There was no evidence of fire.

Employees C and O suited up in coveralls and face masks entered the plutonium handling area carrying a portable beta-gamma survey meter. The 16-inch plastic pouch had been dislodged from Glove Box S-202 and was found lying on the floor along with the sphincter cans, plastic vials, Kimwipes, and fuel powder from the glove boxes. The employees closed the open 16-inch pouch port with the port cover and exited. Subsequently, the facility personnel were formed into de-

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contamination teams, they suited up, and entered the plutonium handling area to remove the loose radioactive debris into waste barrels and to seal off the adjacent areas to prevent the spread of contamination.

Surveys were made of the entire Pawling Plutonium Facility, and its environs to assess the extent of contamination. The level of airborne plutonium activity in the Research Area shortly after the incident was 1×10^{-5} μC/ml of air. The survey of the Research Area gave alpha contamination ranging from about 500 cpm to above the sensitivity of the meter of 500,000 cpm. The instrument survey results taken prior to any cleanup are shown in Table 1 of Appendix A. The results of a smear survey of the Research Area ceiling heat baffles and wall surfaces indicated that these surfaces were not significantly contaminated.

All other areas of the Pawling Plutonium Facility shown in Fig. 1 of Appendix A, with the exception of the Research Area and the emission spectrograph area, have been surveyed and found to be uncontaminated. A survey at the entrance of the emission spectrograph area shown in Table 1 of Appendix A indicated that the area was not significantly contaminated, however, a thorough survey will not be attempted until the radioactivity level in the Research Area has been further reduced.

Continuous monitoring of the Plutonium Facility exhaust stack and a survey of the Pawling site shows that no contamination was released to the environs. The contamination was contained, since the plutonium handling area operates at a negative pressure with respect to the environment, and because all lines that exhaust to the environment are protected by absolute filters.

A decontamination effort was organized and is in process. As of January 24, 1972, the smearable contamination was reduced to 2000 cpm maximum, and the fixed contamination to 27,000 cpm maximum. The airborne radioactivity was within acceptable limits of 2×10^{-13} μC/ml. An extensive monitoring program has been in effect since the incident occurred. Continuous air samples are collected in the plutonium handling area daily and outside of the Plutonium Facility on a weekly basis. The plutonium handling area is surveyed frequently. Bioassay samples (urine and nasal smears) are taken on a continuing basis. A continuing monitoring program of the site area is in progress to assure that plutonium is not released to the environment.

3. PERSONNEL EXPOSURES

No one was exposed to airborne concentrations of plutonium during the excursion since no personnel were present in the Laboratory at that time.

During subsequent inspection, cleanup and decontamination activities, however, concentrations in excess of the values listed in 10 CFR 20, Appendix B, Table 1, existed in the Research Area (see Table 2 of Appendix A of this report). Fifteen employees have been engaged in these activities for various periods of time. Exposures of the employees ranged from 0.01 × MPC to 1.3 × MPC for two consecutive seven-day periods of exposure as shown in Table 6, Column 2 of Appendix A. Exposures were calculated using the average concentration data of Table 2, the working time data of Table 5, and the respiratory protection factor for full-face respirators of 100 listed in 10 CFR 20, Appendix E. These estimates of exposure are felt to be conservative in view of the results of the nasal smear and bioassay program, as given in Appendix A.

Had they not worn full-face respirators, these personnel would have been exposed to from 1 ×MPC to 130 ×MPC for two consecutive seven-day periods. The theoretical exposure data are shown in Table 6, Column 1 of Appendix A. These theoretical exposures without full-face respirators were calculated using the average concentration data of Table 2 and the working time data of Table 5.

Each employee involved in decontamination was fully protected with special clothing described in Table 4 of Appendix A.

4. PROBABLE CAUSE

The three analytical chemistry glove boxes involved in the incident are identified as S-202, S-203, and S-204 in Fig. 1. Each one of the boxes is approximately 3 feet wide × 4 feet long × 3 feet high, and they are interconnected with 16-inch diameter tunnels. The 16-inch diameter ports of the end face of S-202 and S-204 are covered with a polyethylene bag used for pouching.

These three glove boxes draw air from inlet lines that are connected to a common manifold which draws laboratory room air. Each glove box inlet line is protected by an absolute filter.

The air exhaust lines from each glove box exit into a common manifold. The exhaust manifold is maintained at about -0.8-inch water column pressure differential by the Facility exhaust system. Air exits each glove box through its absolute filter into the exhaust manifold, then passes through a second absolute filter located in the Gas House, before exiting into the stack and finally to the environment.

Glove Box S-202 contains equipment for the determination of oxygen-to-metal (O/M) ratio in uranium-plutonium oxides by the gravimetric, oxidation-reduction procedure: a resistance heated furnace with sample retort, and a balance. In this procedure the weighed samples are placed in the retort and oxidized in air at 880°C, and then reduced with a reducing gas (normally 6% H_2-helium) at the same temperature. The reducing gas enters from a cylinder outside the box, passes through the furnace, and exits into the box at a point adjacent to the exhaust filter. The hydrogen content of gas mixtures used for this analysis is certified at Gulf United prior to use. The analysis of the mixture used prior to the incident was verified by an independent testing laboratory.

Glove Box S-203 contains equipment for the determination of carbon by the combustion-gravimetric technique: an induction-heated furnace and retort. The generator for the furnace and the gas absorption train are outside the glove box. The combustion gas (oxygen) enters from a bottle outside the box, passes through the furnace in the box, the absorption train outside the box, and vents back into the box line to exit through an exhaust filter.

Glove Box S-204 contains equipment used for the determination of moisture in uranium-plutonium oxides: standard analytical equipment using a P_2O_5 cell.

Only the O/M analysis setup in Box S-202 was in use at the time of the incident. UO_2-PuO_2 samples were being heated up in the box atmosphere to be oxidized. The reducing gas was shut off.

The reducing gas normally used for the reduction step of the O/M analytical procedure is 6% H_2-He. On December 27, 1971, however, this gas was replaced with a cylinder of 8.2% H_2-Ar. According to Bureau of Mines Bulletin 503, "Limits of Flammability of Gases and Vapors," a hydrogen-helium mixture having less than 8.7% hydrogen will not burn when mixed in any proportion with air. Although such limit mixture data are not available for the case of argon as the diluent, data are presented in this reference which indicate: (1) hydrogen-nitrogen mixtures having more than 5.7% hydrogen will burn when mixed with air; and (2) the replacement of nitrogen with argon tends to decrease the limiting hydrogen concentration. Thus, while the 6% H_2-He mixture is "always safe," the 8.2% H_2-Ar forms a range of flammable mixtures with air. Using the nitrogen data in Bulletin 503, the flammable compositions range, as a minimum, from 48% of the argon-hydrogen mixture to 76% of the argon-hydrogen mixture.

On December 27 and December 28 the O/M procedure was conducted, and at the time of the incident on December 29, the procedure had been started. In addition, on December 28 a carbon analysis was conducted in the adjacent box, and the oxygen* used in that procedure exited into the Box S-202. The time record and flow rate information were reviewed to ascertain the quantities of the various gases which, in addi-

*Bulletin 503 data indicate introducing oxygen broadens the flammability range.

tion to make up air, were introduced into the Chemistry Box Line. These are given below.

Date	Time	Gas	Estimated Total Volume, ft^3
12/27		8.2% H_2-Ar	36
12/28	From 11:00a.m to 4:15 p.m.	Air	20
		8.2% H_2-Ar	15
		Oxygen	4
12/28 12/29	From 4:15 p.m. to 8:45 a.m.	8.2% H_2-Ar	89

Since the boxes are maintained at substantially constant pressure, a portion of each of these constituents left the box through the exhaust filter. However, by the time of the incident, the box atmosphere had evidently become sufficiently enriched in the hydrogen mixture to fall in the flammable composition range.

On December 29 the O/M furnace was turned on at approximately 11:30 a.m. The temperature of the furnace was observed to be approximately 400°C at 12:00 p.m. The normal heating rate of this furnace is such that a temperature in the range of 500 to 600°C would have been attained at 12:35 p.m., the time of the incident. This evidently ignited the mixture, causing the pressure increase which dislodged the pouch on the north end, allowing plutonium-bearing material to become airborne.

Assuming a composition resulting from mixing the gases introduced into Box S-202 from December 27, 1971 to the time of the incident, the maximum (adiabatic) pressure rise associated with burning of the mixture is calculated to be 35 psi. This exceeds the pressure rise required to dislodge the pouch.

The O/M furnace had also been operated with the argon-hydrogen mixture on the previous day, December 28, 1971, without incident. At the time that the furnace was shut off on that day (4:15 p.m.) however, an estimated total of 51 ft^3 of the 8.2% H_2-Ar mixture had been introduced into the box, which was not sufficient to produce a box atmosphere in the flammability range.

5. CORRECTIVE ACTION

The following steps have been taken to prevent recurrence of this type of incident:

1. The reducing gas used in the O/M analytical procedure in Box S-202 will be restricted to 6% H_2-He. Use of 8.2% H_2-Ar in this procedure will be prohibited.
2. No substitution of another diluent for H_2 nor any increase in H_2 concentration will be permitted without a prior feasibility study documenting the safety of the proposed change.
3. The tunnel between the O/M box (S-202) and the carbon analysis box (S-203) will be permanently closed. No oxygen will be vented into Box S-202.
4. All other operations in the Plutonium Laboratory involving the use of either combustible gas or oxygen will be reviewed to prevent similar occurrences.
5. All stored plutonium-bearing powders will be kept in closed containers.

APPENDIX A – SURVEY RESULTS

INSTRUMENT SURVEY RESULTS

An instrument survey was taken of the Plutonium Facility Research Area on December 30, 1971 prior to any cleanup operation. The results of this survey are shown in Table 1. The survey location points are shown in Fig. 1.

RESTRICTED AREA AIR SAMPLE RESULTS

Air samples* are collected at five fixed locations in the Research Area of the Plutonium Laboratory and analyzed daily. Results of these samples covering the period December 29, 1971 (day of incident) until January 19, 1972 are shown in Table 2.

UNRESTRICTED AREA AIR SAMPLE RESULTS

Air samples are collected in the Plutonium Laboratory exhaust stack and at four outdoor locations and analyzed. Results of these samples covering the period of December 29, 1971 to January 21, 1972, are shown in Table 3. The locations of the air samplers are shown in Fig. 2.

PROTECTIVE GEAR

Each individual involved in the decontamination of the Plutonium Laboratory Research Area is attired as described in Table 4.

WORKING TIMES

Table 5 shows the time that each individual worked in the Plutonium Laboratory Research Area during the period of 12/29/71 to 1/11/72. This table is used for evaluation of exposures to airborne radioactive material.

PERSONNEL EXPOSURES

Personnel exposures to concentrations of plutonium in air for two 2-week periods beginning with the date of the incident are shown in Table 6. Exposures were calculated using data from Tables 1 and 5, and both with and without a respiratory protection factor of 100 (10 CFR Part 20, Appendix E).

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*Continuous air sampler using Gelman glass fiber filter paper. Samples were measured for activity on a Nuclear Measurements Corporation Gas Proportional Counter, Model PC-3T. All samples were counted immediately after collection.

NASAL SMEAR RESULTS

Nasal smears are collected from each individual upon leaving the Research Area of the Plutonium Laboratory. The smears are counted on an NMC Model PC-3T gas proportional counter. The highest individual smear was 6 cpm.

URINALYSIS RESULTS

Daily urine samples are collected for each individual working in the Research Area of the Plutonium Laboratory. The samples are submitted to an independent laboratory for analysis.

Results reported for the period December 29, 1971 to January 4, 1972, show no concentrations in excess of 0.11 ± 0.05 dpm Pu/sample.

TABLE 1 — INSTRUMENT SURVEY* DATA FOR RESEARCH AREA OF PLUTONIUM FACILITY PRIOR TO CLEANUP ON DECEMBER 30, 1971

Research Area Section (See Fig. 1)	Location of Survey and Activity in cpm			
	Floor	Top of Glove Box	Wall to 6-ft Level	Top of Pipes (13-ft Level)
1	50,000	50,000	500	500
2	50,000	2,500	< 500	2,000
3	25,000	25,000	500	—
4	50,000	50,000	< 500	—
5	40,000	1,000	—	10,000
6	50,000	1,500	< 500	500
7	>500,000	500	<500	50,000
8	40,000	50,000	1000	40,000
9	>500,000	1,500	1500	750
10	500	—	—	—

*Instrument survey taken with an Eberline Instrument Corporation PAC-4G Alpha Counter.

TABLE 2 — AVERAGE AIR SAMPLE SURVEY DATA FOR RESEARCH AREA OF PLUTONIUM FACILITY

Date	Concentration, μC Pu/ml Air
12/29/71	$1.0 \times 10^{-}$
12/30/71	7.5×10^{-11}
12/31/71	3.0×10^{-9}
1/3/72	3.1×10^{-10}
1/4/72	8.9×10^{-11}
1/5/72	1.2×10^{-10}
1/6/72	2.5×10^{-11}
1/7/72	6.4×10^{-11}
1/10/72	1.9×10^{-12}
1/11/72	3.7×10^{-12}
1/12/72	9.0×10^{-12}
1/13/72	7.0×10^{-12}
1/14/72	2.0×10^{-12}
1/17/72	5.0×10^{-13}
1/18/72	1.8×10^{-12}
1/19/72	3.2×10^{-12}

TABLE 3 — ENVIRONMENTAL AIR SAMPLE SURVEY DATA

Period Ending on Date	Concentration, μC Pu/ml Air				
	Stack	Location A	Location B	Location C	Location D
12/29/71	3.4×10^{-15}	—	—	—	—
12/31/71	1.2×10^{-16}	3.2×10^{-15}	7.2×10^{-15}	1.3×10^{-15}	1.1×10^{-14}
1/7/72	$<1.0 \times 10^{-16}$	2.6×10^{-15}	2.2×10^{-15}	3.5×10^{-15}	2.2×10^{-14}
1/14/72	$<1.0 \times 10^{-16}$	1.5×10^{-14}	3.16×10^{-15}	1.5×10^{-15}	$<1.0 \times 10^{-16}$
1/21/72	5.2×10^{-15}	1.1×10^{-14}	1.66×10^{-14}	8.3×10^{-15}	4.5×10^{-14}

TABLE 4 — DESCRIPTION OF PROTECTIVE GEAR WORN BY GULF UNITED PERSONNEL DURING DECONTAMINATION

1. Underclothing (socks, T-shirts, shorts).
2. Coveralls with openings sealed by masking tape.
3. Surgeon's gloves sealed to coveralls with masking tape.
4. Lab shoes covered with plastic bag, and openings sealed with masking tape.
5. Acid suit plastic pants and jacket, with openings sealed (including vent holes in jacket) with masking tape. Plastic boot over plastic bag covering shoes, with openings sealed by masking tape.
6. MSA full-face mask with No. 84305 filters.*
7. Neoprene gloves over surgeon's gloves, with glove opening sealed to plastic acid suit by masking tape.
8. Surgeon's cloth cap on head.
9. Plastic acid suit hood over cap.
10. Clear plastic bag enclosing mask and hood. Plastic bag sealed to plastic acid suit by masking tape.

*Filtration efficiency of 99.98% for 0.3-micron DOP smoke.

TABLE 5 — WORKING TIME OF ALL EMPLOYEES IN THE RESEARCH AREA (minutes)

Employee	12/29/71	12/30/71	12/31/71	1/3/72	1/4/72	1/5/72	1/6/72	1/7/72	1/10/72	1/11/72
A	105	80	130	135	120	90	110	—	150	110
B	195	—	—	135	120	145	110	145	145	155
C	90	80	105	101	100	—	105	95	—	145
D	—	95	85	115	140	125	105	130	120	100
E	90	—	—	—	—	—	—	—	—	—
F	105	—	90	70	110	110	110	95	—	—
G	90	105	125	—	100	120	120	100	120	155
H	—	100	85	60	—	105	105	95	115	—
I	—	95	90	120	—	145	95	135	140	190
J	90	—	—	—	—	—	—	—	—	—
K	105	80	—	65	115	110	90	145	90	—
L	—	—	—	85	115	105	120	100	105	90
M	—	—	—	85	140	60	95	125	120	90
N	—	80	120	97	—	—	—	110	120	—
O	5	—	—	—	—	—	—	—	—	—

TABLE 6 — EMPLOYEE EXPOSURES

Employee	Concentration, μC Pu/cc Air			
	First Week		Second Week	
	Without Protection Factor	With Protection Factor	Without Protection Factor	With Protection Factor
A	2.3×10^{-10}	2.33×10^{-12}	7.29×10^{-12}	7.29×10^{-14}
B	1.72×10^{-10}	1.72×10^{-12}	8.6×10^{-12}	8.6×10^{-14}
C	1.90×10^{-10}	1.90×10^{-12}	6.36×10^{-12}	6.36×10^{-14}
D	1.63×10^{-10}	1.63×10^{-12}	1.22×10^{-9}	1.22×10^{-13}
E	1.88×10^{-10}	1.88×10^{-12}	—	—
F	2.13×10^{-10}	2.13×10^{-12}	7.54×10^{-12}	7.54×10^{-14}
G	2.5×10^{-10}	2.5×10^{-12}	1.01×10^{-11}	1.01×10^{-13}
H	1.97×10^{-10}	1.97×10^{-12}	1.11×10^{-11}	1.11×10^{-13}
I	2.21×10^{-10}	2.21×10^{-12}	1.21×10^{-11}	1.21×10^{-13}
J	1.88×10^{-10}	1.88×10^{-12}	—	—
K	7.38×10^{-11}	7.38×10^{-13}	1.28×10^{-11}	1.28×10^{-13}
L	3.84×10^{-11}	3.84×10^{-13}	9.27×10^{-12}	9.27×10^{-14}
M	4.08×10^{-11}	4.08×10^{-13}	7.48×10^{-12}	7.48×10^{-14}
N	1.65×10^{-10}	1.65×10^{-12}	3.0×10^{-12}	3.0×10^{-14}
O	$2. \times 10^{-12}$	$2. \times 10^{-14}$	—	—

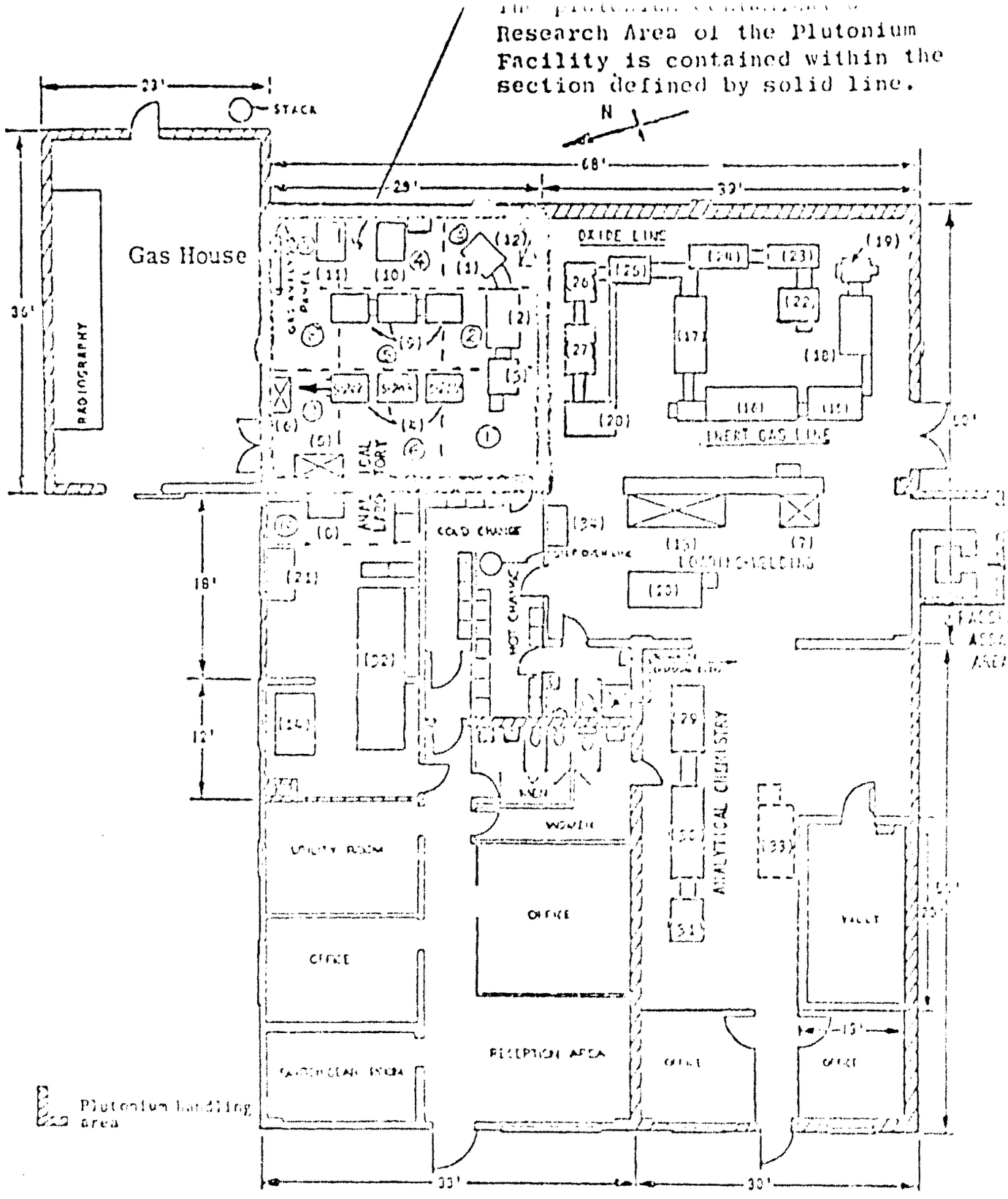

Fig. 1 — Layout of the Gulf United Nuclear Fuels Corporation Plutonium Facility, Pawling, N. Y.

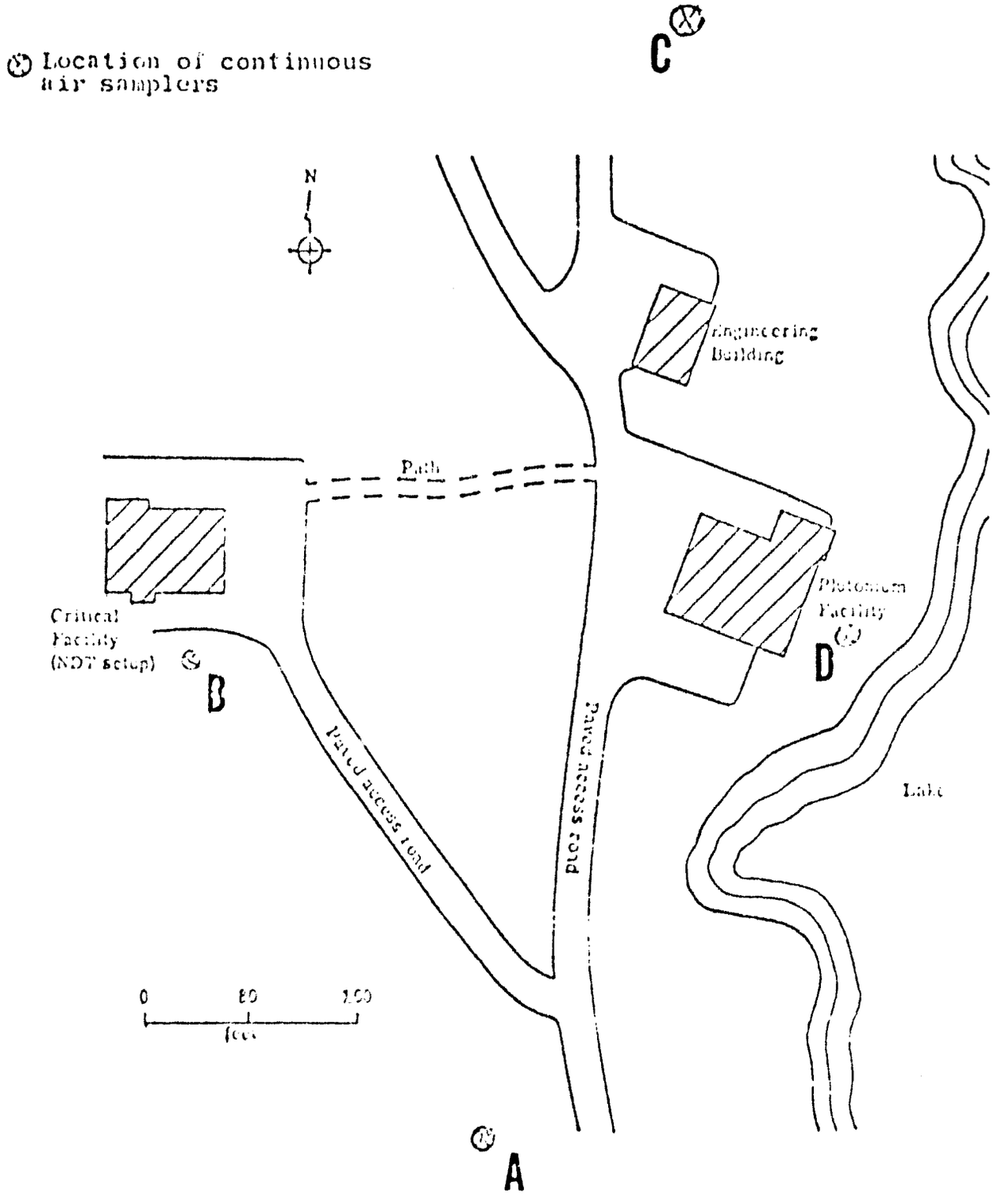

Fig. 2 — Site Plan – Pawling Facility

GULF UNITED
NUCLEAR FUELS CORPORATION

P. O. Box 608
New Haven, Connecticut 06503
203-777-7641

In Reply Refer To: RA:LM-73-42

MAR 28 1973

Mr. Leland C. Rouse, Chief
Fuel Fabrication and Reprocessing Branch
Directorate of Licensing
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: Section 800

Reference: AEC License No. SNM-871, Docket 70-903

Dear Mr. Rouse:

Attached is Section 800 to the referenced license as part of Gulf United Nuclear Fuels Corporation's revision to the renewal application for SNM-871. This Section is part of the demonstration portion of the license and describes the Eastview Facilities of the Elmsford Site. It shows how the technical specifications of Sections 300 and 400 were used to evaluate and establish the safety of these facilities as they now appear.

Sincerely,

Robert E. Kropp
Licensing Manager

REK:fg

A/105

9103200400

REVISED SUBSECTION NUMBERS

Revised Renewal Application	Replaces In Renewal Application LIC-64-3 (Revised)	Replaces In Approved License LIC-64-2 (Revised)
Subpart 811	New	--
Subpart 812	New	--
Subpart 813	Section 11 - pp. 73-74 Appendix B - pp. 109-110	Section 11 - pp. 57-58 Appendix B - pp. 84-85
Subpart 814	Appendix B - pg. 111	Appendix B - pp. 84-85
Subpart 815	Appendix B - pg. 111	Appendix B - pp. 84-85
Subpart 821	New	--
Subpart 822	Section 2 - pg. 13 Section 4 - pg. 19	Section 2 - pg. 10 Section 4 - pg. 18
Subpart 823	Section 2 - pg. 14 Section 4 - pp. 20-21	Section 2 - pg. 10 Section 4 - pg. 18
Subpart 824	Section 2 - pg. 13	Section 2 - pg. 10
Subpart 831	Appendix D - pp. 159-160	--
Subpart 832	Appendix D - pp. 160-161	--
Subpart 833	Appendix D - pg. 161	--

Date: MAR 2 8 1973

Rev: 0

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800. Scope of Section

This Section is one of the demonstration portions of the license. It shows how the standards and criteria of Sections 300, 400, 500, 600 and 700 were used to evaluate the nuclear criticality safety, radiation safety, nuclear materials management, emergency planning and shipping and receiving functions of the facilities as they now appear. These demonstrations indicate how the technical specifications will be used to perform evaluations of future changes and modifications.

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Section 800 - Eastview Facilities

Subsection 810 - Storage

SUBPART 811	OUTSIDE STORAGE
SUBPART 812	INSIDE STORAGE
SUBPART 813	BUILDING 2 VAULT
SUBPART 814	BULK STORAGE OF LOW-ENRICHMENT MATERIALS
SUBPART 815	STORAGE OF LOW-ENRICHMENT FUEL RODS

ense: SNM - 871 Docket: 70-903 Section: 800 Subsection/Subpart: 810

ject: EASTVIEW FACILITIES - Storage - Table of Contents

ed: MAR 28 1973 Supersedes: 5/30/72 Approved: Page 1 of 1

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811. Outside Storage

SNM bearing materials may be stored outside the buildings of the Eastview Facilities within the fenced-in area if the SNM is in a shipping container. Arrays of containers will be stored as described in Subsection 703. Specifically, the following limitations will be employed:

1. Containers will be sealed and not opened in outside areas. Prior to opening, containers will be taken into a process, shipping-receiving or storage area.

2. Containers will be labeled such that their contents can be readily determined. These labels will be weather proof and will either list or reference appropriate internal documents which list the enrichment, quantity of SNM and other pertinent process and health and safety information.

3. SNM bearing liquids will not be stored outside. Acids and corrosive materials will be stored in containers which also meet the criteria of Title 49 CFR.

4. Containers will be stored on pallets to provide elevation off the ground.

5. During periods of high winds, containers will be either moved inside or secured by rope, chain, etc. to the ground or to building walls.

6. Containers and outside storage arrangements will be checked during routine inspections as per Subsection 207.

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812. Inside Storage

1. SNM may be stored in buildings in specified locations, in shipping containers. Arrays of containers will be stored as described in Subsection 703.

2. After unloading from shipping containers, SNM will be stored in storage areas or devices described in this Subsection.

3. In-process storage devices may be placed throughout the buildings to retain SNM during processing or between process steps. These devices are metal racks or concrete bunkers which provide spacing between safe cross section metal boxes or ports, or safe piece count batches.

813. Building 2 Vault

1. Description of Vault

The vault location is shown on Figure 7. It contains the Rod Storage Array, Bulk Storage Array and Archive Storage Array. Details of these storage arrays are shown in Sketch 813-I.

1.1 Rod Storage Array

The storage array for U-Pu rods is installed on the south wall of the Building 2 vault. It consists of a rack for storage of a single row of vertically disposed rods constructed of 1/2-in. steel plate, 36-in. high x 96-in. wide (in two sections). Two horizontal rows of pipe straps are fastened to each section. The rows of pipe straps are 27 in. apart and spaced to accommodate 1 1/4-in. pipe on 3-in. centers.

The rod storage thus consists of a flat array forming a slab of effective thickness $\leq$ 1.5 in. of fixed geometry.

1.2 Bulk Storage Array

The bulk storage array, installed on the north wall of the Building 2 vault, is designed for the safe storage of fissile materials of any enrichment, chemical form, or degree of moderation.

It consists of a fixed array of receptacles each 5 1/4-in. deep x 5 1/4-in. wide x 11 in. high with a horizontal surface-to-surface separation of 12-in. and a vertical surface-to-surface of 13 in. The rack is constructed of 16-gauge steel with a face plate which blanks out all areas except for the openings. A rear plate is bolted to the wall of the vault. Material is stored in cylindrical metal containers, limited to one container per receptacle.

1.3 Archive Storage Array

The archive storage array is constructed on the west wall of the Building 2 Vault. It is designed for the storage of low enrichment (not exceeding 5 w/o U-235) rods with a maximum fuel length of 13 ft. Further details are shown on Sketch 813-II.

It will consist of a series of brackets welded to a vertical support which is attached to the wall. The rods will be supported horizontally and the overall dimensions of the array will not exceed 3-1/2" in thickness and 30" in height. Rods will be strapped or clamped into individual brackets to preclude their falling out.

1.4 Criticality Alarm System

This unit is mounted on the wall of the lab and is a gamma detector utilizing two Amperex 90NB-3 Geiger-Mueller tubes. The unit is capable of accepting three additional remote G-M detectors and these detectors are located throughout the facility. (See Fig. 7.)

This system satisfies the requirements of 10 CRF 70. A rise in the gamma background, as would be experienced during an accidental criticality. causes the alarm to sound, which requires the immediate evacuation of the area.

The criticality alarm has five rechargeable nickel-cadmium cells. During normal operation, the batteries are maintained fully charged by a 200-milliampere trickle charge from the low-voltage rectifier circuit. They begin to discharge only when a-c power fails. The operating time for the unit, if there is a power failure, is 10 hr under high limit alarm conditions, and 90 hr under nonalarm conditions.

2. Nuclear Criticality Safety

2.1 Rod-Storage Array

The maximum safe loading of each rod can be computed on the basis of the homogenized slab corresponding to 3-in. spacing between adjacent rods and a maximum 1 1/2-in. thickness. From Fig. 4 of TID-7016, the maximum U-235 density for an isolated 1 1/2-in. thick slab with full reflector is 1.57 kg/liter, or 1.57 g/cm^3. To convert this to permissible loading per foot, consider a unit cell containing a 1-ft length of rod. Its volume is 12 in. x 1 1/2 in. x 3 in. = 54 $in.^3$ = 0.866 liter. Hence, making the conservative assumption that one gram of Pu-239 is equivalent to 4 grams of U-235, the permissible loading of the rod is:

(Mass U-235 + 4.0 x Mass Pu-239) $\leq$ 1.57 kg/liter x 0.886 liter/ft.

(Mass U-235 + 4.0 x Mass Pu-239) $\leq$ 1.39 kg U-235 equivalent/ft.

This restriction will ensure the safety of the isolated rod storage array.

2.2 Bulk Storage Array

Amended Table IV of TID-7016 indicates that for H/U-235 ratios greater then 20, a volume limit of 3.6 liters is always safe, and that for H/U-235 ratios less than 20 a mass limit of 3.6 kg U-235 is always safe. Therefore, a mass limit of 3.6 kg of U-235 or equivalent, and a cylindrical container volume limit of 3.6 liters are established for each storage receptacle.

The bulk storage array satisfies the requirement of Table IV of TID-7016 as cited above, and Table V of TID-7016 indicates that an isolated plane array of such units is safe for any number of units, provided that:

1. Spacing between unit containers is at least 8 in. surface-to-surface; and

2. Minimum center-to-center spacing of units is at least 16 in.

Since the bulk storage array satisfies both conditions, the array is safe.

2.3 Archive Storage Array

The array is analyzed assuming that it is an isolated zone as would be the condition in case of flooding.
Use is made of a series of diffusion theory calculations performed for the analysis of fuel rod storage in the Critical Facility and reported in NDEO-701 dated June 1966 (reference AEC License R-49, Docket 50-101).

Values of K_{eff} were calculated for semi-infinite fully reflected slabs of unclad fuel rods containing oxide fuel enriched to 5 w/o U-235 in U. A series of calculations in which rod diameter and water to fuel ratio were varied led to the conclusion that an infinite slab thickness of 9.75 cm was safe (i.e., $K_{eff} < .98$).

Performing a buckling conversion (using a reflected extrapolation length of 6.6 cm) to a 3.5 in. thick 30-inch wide slab (assuming infinite length) results in a K_{eff} value for the fully reflected system of 0.92. Thus, the proposed storage is safe against flooding.

The Keff value for the bare system was calculated by assuming the same k_∞ and M^2 values as for the reflected system and using a bare extrapolation length of 3.2 cm to calculate buckling values. The result for the optimally moderated system of unclad rods was 0.64.

2.4 Interaction

The K_{eff} values for all the units in the vault have been calculated under the assumption of double batching (where appropriate) and no reflector. Calculations were performed by a one-dimensional discrete ordinates code (DTF) except for the archive array which was calculated as described above. The results are as follows:

Bulk Storage Unit (double batched):	0.66
Rod Storage Array (double batched):	0.59
Archive Array:	0.64

According to Fig. 26 of TID-7016 (Rev. 1) an allowable solid angle for a k_{eff} value of 0.66 is 2.4 steradians.

Solid angles have been calculated by the "point-to-plane" method listed on page 35 of TID-7016 (Rev. 1) and are:

Case No.	Solid Angle Subtended By	At the Center of	
1	Bulk Storage Array	Archive Array	.14 steradians
	Rod Storage Array	Archive Array	.32 steradians
		Total	.46 steradians
2	Archive Storage Array	Rod Storage Array	.56 steradians
3	Other Bulk Storage Units	Central Bulk Storage Unit	.81 steradians
	Archive Array	Nearest Bulk Storage Unit	.19 steradians
		Total	1.00 steradians

The separation of the Rod Storage Array and the Bulk Storage Array meets the isolation criteria listed on page 27 of TID-7016 (Rev. 1) and thus may be considered as isolated from each other.

The maximum solid angle (1.0 steradian) is smaller then the allowable solid angle (2.4 steradians) so the storage arrangement is safe.

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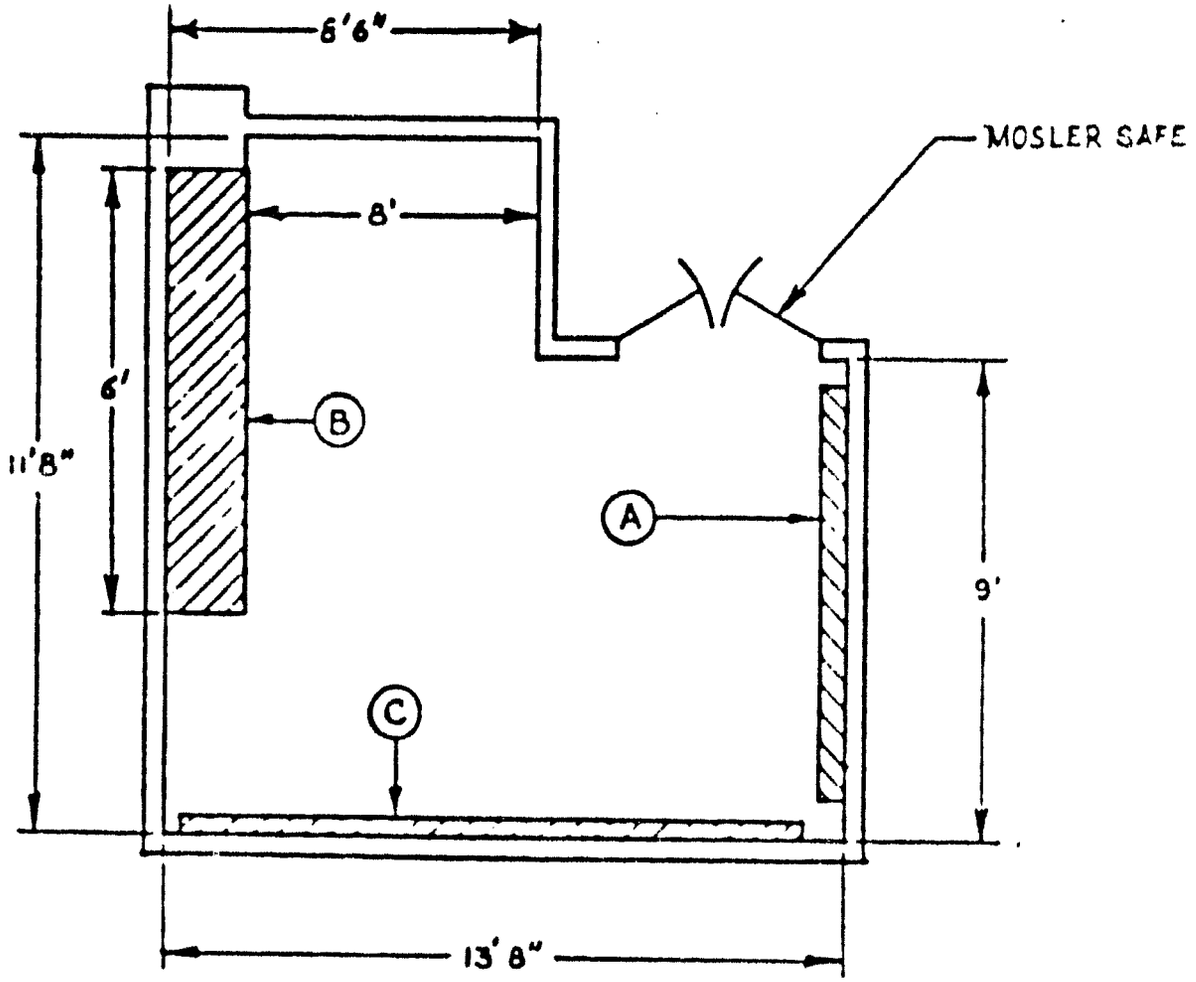

(A) — ROD STORAGE AREA

(B) — BULK STORAGE AREA

(C) — ARCHIVES STORAGE AREA

N.

BUILDING 2 VAULT

nse: SNM-971 Docket 70-903 Section: 800 Subsection/Subpart 813

ect: Sketch 813-I -- Building 2 Vault

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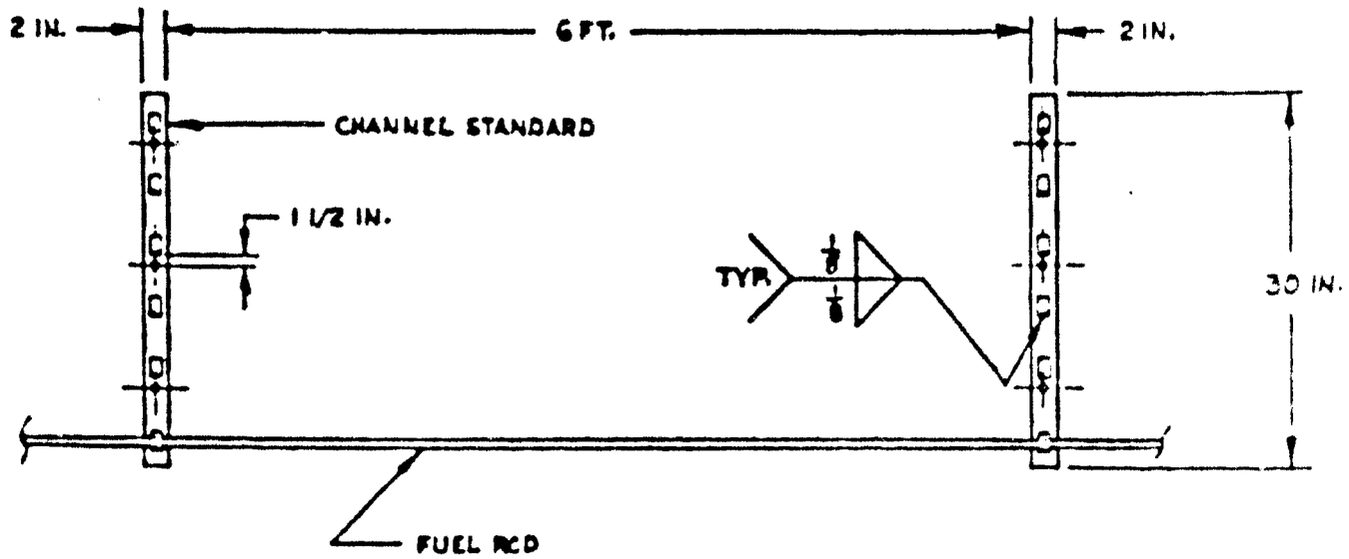

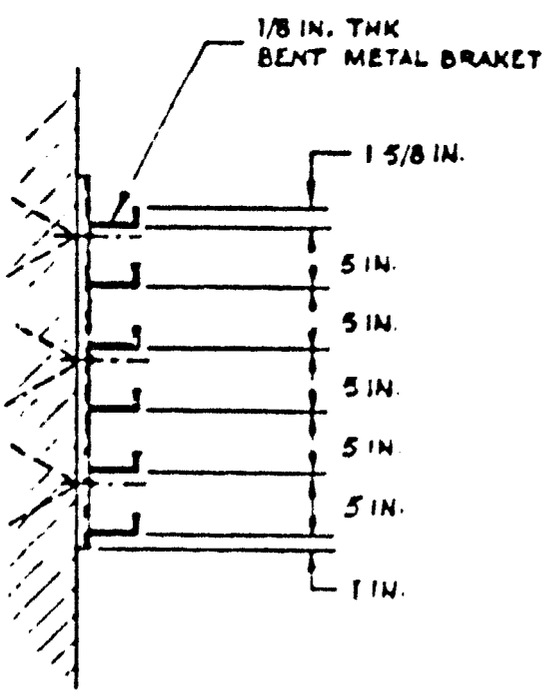

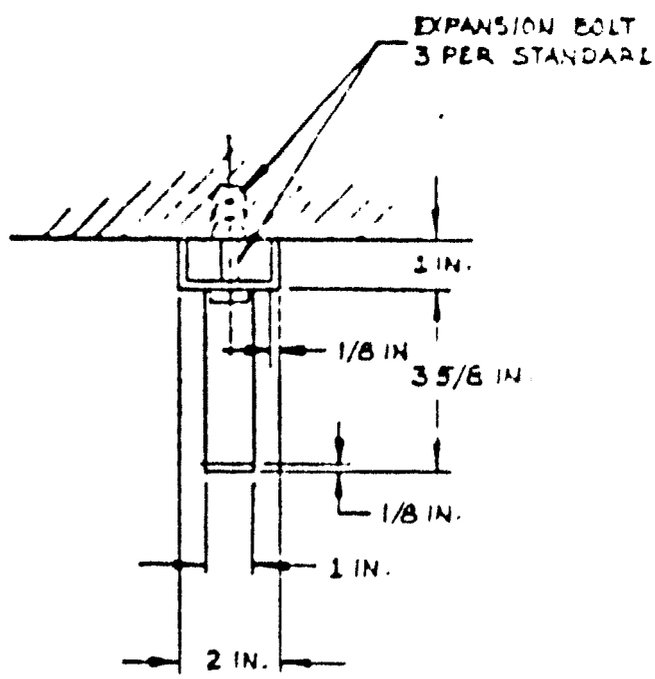

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me: SNM-871 Docket: 70-903 Section: 800 Subsection/Subpart: 813

ect: Sketch 813-II -- Archive Fuel Rod Storage Array

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d: MAR 28 1373 Supersedes: 10/5/72 Approved: _____ Page 1 of 1

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814. Bulk Storage of Low-Enrichment Materials

Special nuclear materials enriched in 5% U-235 or less may be stored in low-enrichment storage arrays to be constructed in the Uranium Fuel Development Laboratory. The types of materials to be stored in those arrays will include powder (primarily uranium oxide), pellets, fuel pellet stacks, and fuel rods.

Uranium oxide powder and pellets will be stored in an array constructed of metal shelving with receptacles sized to accept one metal container. Intermediate spaces will be blanked off to maintain a vertical and horizontal surface-to-surface separation of 2 ft between containers.

Limits for the maximum quantity of UO_2 which may be stored in each receptacle will be established, using Figure 309-XVII for powder and 309-XIII for pellets.

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815. Storage of Low-Enrichment Fuel Rods

Low-enrichment (not exceeding 5 w/o U-235) fuel rods may be stored in a safe geometry array. The rods will be physically secured in a fixed-geometry rack. This rack, loaded with rods, may be contained in a safe thickness of infinite slab not exceeding that given by Figure 309-XV.

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Section 800 - Eastview Facilities

Subsection 820 - Processing

SUBPART 821	INTRODUCTION
SUBPART 822	MATERIAL TECHNOLOGY LABORATORIES
SUBPART 823	URANIUM FUEL DEVELOPMENT LABORATORY
SUBPART 824	CHEMISTRY LAB

821. Introduction

1. The following subsections describe the handling, processing steps and some of the procedural controls used. Individual process steps, equipment, items, support information and safety limitations and considerations, and processing information are described in the appropriate subsections.

2. The Uranium Fuel Development Laboratory (Rooms 31, 33, 34A, 34B, 34C, 35, 39, 40 and 41), the Building 2 Vault (Room 32) and the Change Room (Room 24) as shown on Fig. 7 constitute a Controlled Area where un-clad, bare, exposed and unencapsulated uranium may be processed.

3. The interaction between process operations is not calculated. Generally, each process operation is separated by three feet and identified as a criticality zone. This separation is exclusive of ports or shelves in storage devices. In lieu of determining the interaction between these process operations, a process area limit has been established using a safe surface density of 175 gm U-235 per square ft. The Process Area Limits are:

Facility	Process Area Limit
Material Technology Laboratories	933 Kgs U-235
Uranium Fuel Development Laboratory	1,159 Kgs U-235
Chemistry Laboratory	263 Kgs U-235

Details concerning these limits are set forth in Nuclear Safety Evaluation 821-I. The Possession Limits set forth in Section 100 will not be exceeded regardless of the Process Area Limit.

4. The safe values referenced in the following Subparts will be obtained using The Tables and Figures referenced below:

Safe Value	Table/Figure
Mass	309-I,-II,-III,-XIII,-XVII,-XXI,-XXIII,-XXV
Volume	309-I,-XIV,-XVIII,-XXVI
Diameter	309-I,-XII,-XVI,-XX,-XXII,-XXVIII
Thickness	309-I,-XV,-XIX,-XX,-XXVII
Linear Density	309-III

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I. Determination of Safe Surface Density Value

A safe surface density value of 175 gms U-235 per square foot of floor area was used. Reference Subsection 303.

II. Process Area Limit - Material Technology Laboratories

The Material Technology Laboratories are located in Rooms 17, 18, 19, 20, 23, 25, the Liquid Metals Lab and the X-Ray as shown on Fig, 7. These rooms and labs have a floor area of approximately 5,330 square feet.

Process Area Limit = 5,330 ft^2 x 175 gm U-235/ft^2 = 933 Kgs U-235

III. Process Area Limit - Uranium Fuel Development Laboratory

The Uranium Fuel Development Laboratory is located in Rooms 31, 33, 34A, 34B, 34C, 35, 39, 40 and 41 as shown on Fig. 7. These rooms have a floor area of approximately 6,622 square feet.

Process Area Limit = 6,622 ft^2 x 175 gm U-235/ft^2 = 1,159 Kgs U-235

IV. Process Area Limit - Chemistry Lab

The Chemistry Lab is located in Rooms 21 and 22 as shown in Fig. 7. These rooms have a floor area of approximately 1,500 square feet.

Process Area Limit = 1,500 ft^2 x 175 gm U-235/ft^2 = 263 Kgs U-235

822. Material Technology Laboratories

1. General Considerations

These laboratories are equipped for research and development on liquid and solid metals. Preparation of metallographic specimens, radiography, sterophotography, vacuum melting, hot pressing, powder pressing, welding, and heat treating are performed here. Also, clad fuel rods containing plutonium and uranium may be handled outside the vault.

2. Process Description

2.1 Uranium Fuel Rods

2.1.1 Typical operations will consist of testing procedures applied to a fuel rod which has been loaded in the Uranium Fuel Development Laboratory and sealed at one end. A plug will be placed in the open end as the rod is transported between laboratories.

2.1.2 Typical operations performed on the uranium-containing rods will involve analyses of gases contained in a sealed as-fabricated fuel rod containing uranium compounds only, with gas samples obtained by puncturing the fuel rod.

2.1.3 Testing of unirradiated fuel rods containing uranium only may be handled outside of the Uranium Fuel Development Laboratory. These operations will be performed in the Electron Beam Laboratory (Fig. 7, Room 20) utilizing the welding glove box, and in the Chemistry Laboratory.

2.1.4. In addition, chemical analyses will be performed in the Chemistry Laboratory. The proposed work involves handling of small samples of depleted and enriched uranium compounds, primarily uranium oxides.

2.2 Uranium-Plutonium Rods

2.2.1 The operations performed include: inspection for dimensions and surface condition; X-ray examination for position of the fuel, springs, and penetration of the welds; heating fuel rods to eliminate bubbles in the fuel clad sodium bond; sodium fill inside of a second capsule to fill the interstice space with sodium; welding in a glove box to seal the second clad after sodium filling; and eddy current testing for voids in sodium bonds. When the above operations are performed, the uranium and plutonium are contained in at least one all-welded and inspected container.

3. Nuclear Criticality Safety

Each room will be limited to either one of the safe values listed in Subsection 821.4 or one fuel assembly. For example, the Electron Beam Laboratory (Room 20, Fig. 7) typically handles only one rod at any one time containing not more than 75 gm of U-235 and the Chemistry Lab typically processes not more than 20 gm U-235. Both these amounts meet the limits listed in Table 309-I.

823. Uranium Fuel Development Laboratory

1. General Considerations

This is a separate area which is entered through a cold change and hot change room. It consists of a chemistry and ceramics area, a fuels test lab, a machine shop, a metallography lab, and a high bay materials processing lab. Throughout the area there are special walk-in hoods and fume hoods which are connected to a separate ventilation system. All the hoods connect through this ventilation system to high efficiency filters before the air is passed out of the building. The materials processing laboratory, a high bay area, is maintained under a negative pressure of approximately 1/2 in. of water. This uranium handling area is designed to perform experiments and pilot lot production on the chemical processing of uranium compounds, pelletizing and sintering of uranium compounds, experimental fabrication of cermet fuels, and casting and fabrication of uranium-containing alloys. A detailed layout of this laboratory is available in Fig. 7 (Rooms No. 35, 34A, 34B, 34C, 32, 33, 31, 39, 40, 41). Room No. 32 is the vault, located in the Uranium Fuel Development Laboratory, that is used for the storage of fissile materials.

The Uranium Fuel Development Laboratory will conduct research and development and pilot plant studies, the fabrication of ceramic fuel, the manufacture of cermet fuel, the casting and subsequent working of uranium-containing alloys, the testing of various uranium-containing fuels, and the metallography of various uranium-containing fuels.

A typical program conducted in the Uranium Fuel Development Laboratory involved the development of processes for conversion of UO_2 powder to sintered fuel pellets. The program involved development of the process on a pilot plant scale, and upon completion, the process was scaled up in the production facilities of Gulf United. In the course of the program, a major portion of the facilities and equipment of the laboratory was utilized.

2. Process Description

Typical operations are summarized below:

2.1. Granulation

The initial phase of the program was devoted to development of suitable processes for press feed preparation, including studies of granulation by both wet and dry techniques. In initial experiments, UO_2 powder was mixed with various binders in a liquid-solids blender located in a chemical hood in Room 41. The granulations were characterized by sieving and weighing operations, also conducted in a chemical hood, and finally were dried in a forced draft oven.

In an alternate process, the powders were granulated by compacting in a mechanical press, and the compacts were disintegrated in a mill located in the walk-in hood in Room 41. The product from the mill was classified by a separator, located in the press-feed box in Room 31.

Material produced by either process was passed through a granulator in preparation for pressing; the operation was performed in the press feed box in Room 31.

2.2 Pressing

Granulations produced by these techniques were evaluated by preparation of compacts with a mechanical press. The press, located in Room 31, is fitted with a special enclosure vented into the hot exhaust system.

2.3. Dewaxing

The pressed green compacts were dewaxed by heating to approximately 1650°F in a protective atmosphere. This operation was carried out in an Inconel tube furnace in Room 31. Compacts loaded in molybdenum boats were automatically pushed through the furnace tube following a determined cycle. Special ducts positioned at each end of the furnace were connected to the hot exhaust system. The effectiveness of dewaxing techniques was evaluated by chemical analysis of pellets.

2.4. Sintering

The dewaxed pellets were sintered in a dry hydrogen atmosphere using a molybdenum-wound furnace, also located in Room 31, and connected to the hot exhaust system similarly to the Inconel tube furnace.

2.5. Grinding

Representative sintered pellets were centerless ground in the centerless grinder, located in Room 39, using a water-oil emulsion type of coolant. A special ventilation duct positioned at the grinding wheel and connected to the hot exhaust system prevented any airborne contamination during the operation.

2.6. Inspection

Characterization of ground and unground pellets was conducted by means of density measurements, visual inspection, chemical analyses, and metallographic examination. All operations were carried out in chemical hoods or suitable vented enclosures.

3. Nuclear Criticality Safety

Same as described in Subpart 822.3.

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824. Chemistry Lab

1. General Considerations

The Chemistry Lab is located in Rooms 21 and 22 of Building 2 as shown on Fig. 7.

2. Process Description

The Chemistry Lab at Eastview is equipped for the following activities:

2.1 Qualitative and quantitative analyses using volumetric and gravimetric procedures and techniques of colorimetry, chromatography, atomic absorption, and radiochemistry.

2.2 Trace impurity analysis in nuclear fuels, coolants and alkali metals.

2.3 Environmental health and safety monitoring.

3. Nuclear Criticality Safety

Same as described in Subpart 822.3

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Section 800 - Eastview Facilities

Subsection 830 - Support Activities

SUBPART 831	VENTILATION SYSTEM
SUBPART 832	LIQUID WASTE SYSTEM
SUBPART 833	SOLID WASTE DISPOSAL

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831. Ventilation System

1. The ventilation system is of such a design that the entire laboratory area is maintained at a slight negative pressure. Each hood is exhausted to the ventilation system through an absolute filter. Each item of equipment capable of dispersing particulate matter and contributing to airborne contamination will be enclosed and exhausted to the ventilation system through an absolute filter.

2. A minimum lineal air velocity of 100 ft per minute is maintained at each vented opening. Maintenance of the required air velocity is confirmed by measurement, prior to starting operations involving the particular hood or equipment at the time installation is completed. The air velocity will be checked periodically.

3. The filtering media are roughing filters and absolute filters. The absolute filters are capable of filtering out micron-size particles at high efficiencies. All such filters are given a functional test according to the "Minimal Specification for the Fire-Resistant High-Efficiency Filter Unit," described in the AEC Health and Safety Information issue No. 212, dated 6/25/65. The absolute filters shall meet military specification MIL-F-51068A and "High Efficiency Air Filter Units" UL-586.

4. Hoods are made of various types of material, depending on the type of work permitted in the particular unit. Materials of construction consist of Fiberglas, stainless steel, carbon steel, and transite. Windows consist of Plexiglas and safety glass.

5. Exhaust ducts and filter plenums for the laboratory system are made from galvanized steel. The inside surfaces have been coated with a fire-and chemical-resistant epoxy paint to prevent corrosion by chemical fumes.

6. All components of the ductwork system consist of fire-resistant materials. Since the materials will not support combustion, no analysis of the effects of fire has been made.

7. Filters in the exhaust system are completely sealed in the housings. Changing of this type of filter is accomplished by pulling them out of the housing into plastic bags.

8. Prefilters are used in every location to preserve the absolute filters.

9. Maximum allowable pressure drop across a filter is 3-in. W.G. Each filter housing has pressure taps permanently mounted so that pressure drop measurements may be made.

10. Filters to be used in the plenum are DOP tested at ORNL prior to use. DOP testing of installed final filters will be performed.

11. The types of work permitted in the hoods do not impose any live loads. The static loads consist of only the weight of chemical laboratory types of equipment.

12. The materials of construction are fire-resistant and corrosion-resistant.

13. Unclad plutonium is not permitted anywhere in this facility. Inasmuch as only uranium will be used, a maximum permissible surface contamination level for the hood enclosures has not been established.

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832. Liquid Waste System

1. The laboratory is equipped with a liquid waste system which is separate from the cold drainage system.

2. Hot liquid wastes are collected alternately in hold-up tanks, each of 2000-gallon capacity, which are located below grade in a hot waste pit adjacent to Building 2.

3. Liquid waste collected in each tank is evaluated for contamination by radiological analysis.

4. If analysis indicates an acceptable low level of contamination, the contents of the tank, after filtration, may be dumped into the sanitary sewer. No dumping may be done without approval of the NIS Representative.

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833. Solid Waste Disposal

1. Waste material generated in the laboratories consists primarily of solids. Small quantities of liquid waste which are accumulated may be converted into a solid form or absorbed into special absorbents for disposal. Waste consisting of special nuclear material will be shipped to commercial facilities for recovery or disposal, as prescribed by AEC regulations.

2. Solid waste generated at Elmsford is stored in the Waste Storage Area after the drum has been filled and the lid sealed. During the filling of the drum, the waste is stored inside the laboratory.

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JOHN D. LEONARD, JR.
VICE PRESIDENT - OFFICE OF CORPORATE SERVICES
AND
VICE PRESIDENT - OFFICE OF NUCLEAR

SNRC-1797

MAR 2 1 1991

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Submittal of the 10CFR50.59 Report for the
Period January 1, 1990 Through December 31, 1990
Shoreham Nuclear Power Station - Unit 1
Docket No. 50-322

Gentlemen:

In accordance with the requirements of 10CFR50.59, this letter transmits Enclosure A, Shoreham Nuclear Power Station 10CFR50.59 Report for the period from January 1, 1990 through December 31, 1990. Title 10 CFR Section 50.59 requires that this report list those changes, tests and experiments which did not, by safety evaluation, involve an unreviewed safety question and were completed during the reporting period.

Please be advised that 10CFR50.59 safety evaluations performed in support of Shoreham's System Layup Implementing Packages (SLIPs) program are not included in the enclosed report. The purpose of the lay-up program is to protect from gross deterioration those systems not required to meet Technical Specification requirements or necessary to support minimal plant functions. Station Procedure (SP) S2.001.01, "SNPS Layup Program Description," which has been reviewed by NRC Staff and Region I personnel, provides details on the preparation, contents and approvals of system layup implementation packages (SLIP). 10CFR50.59 safety evaluations associated with the SLIP program have been routinely reviewed by NRC Staff and Region I personnel and these reviews have been documented in various Inspection Reports (e.g., NRC Region I Inspection Report 50-322/90-01).

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The format of the enclosed report is as follows:

SM/SPCN/SEPS No. - In the report, 10CFR50.59 items completed during the reporting period are listed by their Station Modification (SM) number, Station Procedure Change Notice (SPCN) number, or Safety Evaluation Post Settlement (SEPS) number. For convenience of reference, these are each listed separately in ascending order. Associated documents, i.e., Design Output Packages (DOPs), are also given where applicable.

Description of Change - A brief description of the change, test or experiment addressed by the change document.

Summary - The safety evaluation determination that the change, test or experiment does not involve an unreviewed safety question pursuant to the three criteria of 10CFR50.59(a)(2).

Should you require any additional information concerning this submittal, please do not hesitate to contact this office.

Very truly yours,

John D. Leonard, Jr.
Vice President, Office of Corporate Services
and Vice President, Office of Nuclear

MAP/ab
Enclosure

cc: S. Brown
T. T. Martin
B. Norris

SNRC-1797
ENCLOSURE A

10 CFR §50.59 Report

Period From January 1, 1990
Through December 31, 1990

Shoreham Nuclear Power Station
Docket No. 50-322

SM 87-028 (DOP 87-044)

Description of Change

This modification changed the Offsite Dose Calculation Manual (ODCM) software routines in the Radiation Monitoring System (RMS) computer (1D11) which perform the offsite dose calculations. The change allows for accurate calculations when monitors measure less than 20 cpm at RE-42, RE*21, and RE*22.

Summary

I. No. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report has not increased because the change only affects the method of calculation of offsite doses due to gaseous effluent releases from the plant. The modification does not affect the operation of the monitors or any safety function or safety related system.

II. No. A possibility for an accident or malfunction of a different type than any evaluated in the Safety Analysis Report has not been created. This modification does not affect the operation of any safety related system, it merely assures that the monitor's data are used properly.

III. No. The margin of safety as defined in the basis of any Technical Specification is not affected because the detectors only monitor plant effluents and do not affect any plant limiting design conditions.

SPCN 90-0250
SPCN 90-0553
SPCN 90-0714

Description of Change

SPCN 90-0250 revised Station Procedure (SP) 12.006.01, "Station Procedures - Preparation, Review, Approval, Change Review and Cancellation," by authorizing the appropriate Division Managers to assume approval authority for his/her designated procedures. SPCN 90-0553 and 90-0714 revised Station Procedure (SP) 12.001.01, "Index and Organization of Station Operations Manual," by changing "P" approval levels to "D" approval levels, as appropriate. These revisions were made to achieve consistency with Technical Specification Section 6.8, "Procedures and Programs," of Shoreham's operating license NPF-82, as amended.

Summary

I. No. These changes were administrative in nature and do not affect plant operations. Changing the approval authority for procedures has no impact on accidents evaluated in the safety analysis report, nor does it create the possibility for a new accident. Each Division Manager responsible for a given procedure has the qualifications, experience and familiarity necessary to ensure that designated procedures approved by him/her comply with regulatory requirements. Other elements of procedure review, such as ROC review, are not affected by these changes.

II. No. See I above.

III. No. See I above.

SPCN 90-0510

Description of Change

SPCN 90-0510 created a Temporary Procedure (TP) 25.136.01, "Site Characterization of the RX Vessel." The purpose of this temporary procedure is to coordinate the efforts of H.P. Maintenance, Reactor Engineering, and Operations Sections in obtaining exposure data within the reactor (RX) vessel and its internals and provide guidance to allow site characterization of the reactor vessel.

Summary

I. No. The defueled plant condition with fuel stored in the Spent Fuel Storage Pool with the fuel pool gates installed eliminates safety concerns with respect to water level in the reactor vessel, movement of control rods in the reactor vessel, and the bypassing of level and control rod movement interlocks required for safety when fuel is in the reactor vessel.

II. No. See I above.

III. No. See I above.

SPCN 90-0945

Description of Change

SPCN 90-0945 revised Station Procedure (SP) 12.014.05, "Required Reading List," by removing control of the required reading list from the Training Administrative Supervisor and placing the control with individual Sections.

Summary

I. No. The change places the control of distribution of the required reading list from the Training Department to appropriate Plant Staff Section Heads. The change is administrative in nature and unrelated to any accident analysis and does not affect the function or operation of any plant system or equipment.

II. No. See I above.

III. No. See I above.

SPCN 90-1074
SPCN 90-1083

Description of Change

SPCN 90-1074 and SPCN 90-1083 revised Station Procedures (SP) 27.421.01, "RBSVS and CRAC Water Chiller Load Capacity Verification Test," and (SP) 23.421.01, "RBSVS/CRAC Chilled Water," to test the chillers under as realistic conditions as possible and to set chiller maximum load consistent with expected maximum loads in Shoreham's defueled condition, respectively. The load capacity of the M50 chillers was modified to reflect actual heat loads which exist with Shoreham in a defueled/fuel in the Spent Fuel Storage Pool condition.

Summary

I. No. There is no increase in the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report. The probability of a chiller malfunction is actually reduced.

II. No. No physical alterations of plant configuration were made. The modification to load capacity enhances the performance of the chillers.

III. No. See I and II above.

SPCN 90-1172
SPCN 90-1173
SPCN 90-1174
SPCN 90-1177
SPCN 90-1178

Description of Change

SPCN 90-1172 revised Station Procedure (SP) 61.010.01, "Access Controls."

SPCN 90-1173 revised Station Procedure (SP) 62.004.01, "Station Personnel Monitoring Program."

SPCN 90-1174 revised Station Procedure (SP) 62.004.03, "Bioassay Program."

SPCN 90-1177 revised Station Procedure (SP) 61.010.03, "Health Physics Postings and Signs."

SPCN 90-1178 revised Station Procedure (SP) 61.012.01, "Personnel Dose Limits and Guides."

Radiation surveys generated in support of Shoreham's Site Characterization Program have revealed near background measurements throughout the accessible areas of the plant. To reduce the number of personnel that are required to wear dosimetry (TLD/DRD) inside the RCA, the above cited procedures were revised to require dosimetry (TLD + DRD) in only those areas requiring radiation work permits (RWPs). The requirements for access to various areas have been revised to reflect these changes.

Summary

I. No. The changes to the procedures cited above are unrelated to any accident analysis and do not affect the function or operation of any plant system or equipment.

II. No. See I above.

III. No. See I above.

SPCN 90-1176

Description of Change

SPCN 90-1176 revised Station Procedure (SP) 12.014.03, "General Employee Training," to eliminate Visitor Training - Level II and change the requirement for GET II training to be consistent with changes made by SPCN 90-1172 and other associated SPCNs.

Summary

I. No. The changes to the procedure are unrelated to any accident analysis and do not affect the function or operation of any plant system or equipment.

II. No. See I above.

III. No. See I above.

SPCN 90-1406
SPCN 90-1422
SPCN 90-1423
SPCN 90-1424

Description of Change

As described in SNRC-1744 dated August 10, 1990, the Long Island Lighting Company (LILCO) completed a revision to the Shoreham Environmental Qualification (EQ) Program in accordance with 10CFR50.59. This EQ Program revision permitted (1) the deferral of EQ related preventive maintenance tasks and (2) the use of parts not qualified in accordance with 10CFR50.49 in applications where qualified parts would be used to withstand the harsh environmental conditions which could exist at an operating nuclear power plant. The technical justification for the EQ Program revision and safety evaluation were provided in SNRC-1744.

To attain consistency with the revised EQ Program, the following Station Procedures were revised:

1) SPCN 90-1406 and 90-1423 revised Station Procedure (SP) 12.013.01, "Maintenance Work Requests."

2) SPCN 90-1422 revised Station Procedure (SP) 12.015.01, "Preventive Maintenance Program."

3) SPCN 90-1424 revised Station Procedure (SP) 12.020.01, "Environmental Qualification Program."

The revisions addressed the deferral of EQ related maintenance and the use of parts not qualified in accordance with 10CFR50.49 and the record keeping function to track all EQ requirements and the replacement of EQ equipment and/or components.

Summary

The changes to Station Procedures cited above are within the scope of the safety evaluation provided in SNRC-1744.

I. No. Most of the accidents previously evaluated in the safety analysis report cannot occur with the reactor defueled and those that can 1) do not require equipment qualified in accordance with 10 CFR 50.49 to mitigate their consequences, and 2) have consequences that are well within the limitations of 10CFR100.

SPCN 90-1406
SPCN 90-1422
SPCN 90-1423
SPCN 90-1424
(continued)

II. No. A review of USAR Chapter 15, a consideration of the defueled condition, and an assessment of the impact of the EQ Program modification does not reveal any conditions or circumstances that could result in an accident different than those already described in the USAR. There is no need for the EQ equipment while Shoreham is in the defueled, non-operating condition.

III. No. See I & II above.

SPCN 90-1754

Description of Change

SPCN 90-1754 created a new Station Procedure (SP) S2.001.02, "SNPS Decontamination Program," to provide the methods and administrative controls to decontaminate plant systems, equipment and structures. The contamination level of Shoreham's structures, systems and components have been evaluated under the SNPS Site Characterization Program. Those structures, systems, and components found with contamination levels above the acceptance criteria of NRC Regulatory Guide 1.86 are being (will be) decontaminated in accordance with the methods and controls of this procedure. Addendum 1 to the Site Characterization Program will be used to control the radiological surveys as close as practical to the Final Termination Survey Plan as outlined in the Decommissioning Plan submitted by LIPA on December 29, 1990.

Summary

I. No. Program Implementation utilizes a non-destructive chemical decontamination and survey program. The programs will be applied to systems not required for safe shutdown or to mitigate the consequences of an accident in the defueled plant condition. Program implementation is independent and unrelated to applicable accidents evaluated in the USAR in the defueled plant condition. The plant will not be operating during the performance of this procedure. The execution of this procedure is a conservative action taken to reduce contamination in preparation for decommissioning.

II. No. See 1 above.

III. No. See I above.

SEPS 90-004

Description of Change

This change involved the re-assignment of the Radiochemistry/ ALARA and Health Physics Section Head of the Nuclear Engineering Department to the position of Manager, Radiological Controls Division (see SNRC-1711, April 6, 1990 and SNRC-1717, April 27, 1990) and the consolidation of organizational units within the Nuclear Engineering Department (NED) to more effectively utilize NED personnel for activities that are appropriate to the Shoreham Nuclear Power Station (SNPS) defueled condition and the requirements of a full power operating license with the overall functional capability of the Department being retained. Specifically, the following organizational changes were involved:

1. The consolidation of the Reliability/PRA Section with the Safety Analysis Section into a new section entitled Reliability/Safety Analysis Section.

2. The consolidation of the Radiochemistry/ALARA/Health Physics Section with the Nuclear Fuel Engineering Section into a new section entitled Radiation Protection/Fuels Section.

The organizational changes, described above, were reviewed and a safety evaluation was prepared pursuant to 10CFR50.59(a)(i) within the broad premise that plant organization and personnel constitute, in part, "a segment of the facility as described in the safety analysis."

Summary

I. No. The changes are administrative in nature and are unrelated to any accident analysis and do not affect the function or operation of any plant system or equipment. The probability of accidents previously evaluated is not affected by these organizational changes. In Shoreham's non-operating, defueled condition, only the fuel drop accident and the radwaste tank rupture accident apply. The continued technical capability of NED is maintained to comply with license requirements.

II. No. No physical alterations of plant configuration or changes to setpoints or operating parameters were made. The changes only reflect administrative control changes of various functional activities.

III. No. The changes are unrelated to any margin of safety.

SEPS 90-005

Description of Change

As described in SNRC-1743 dated August 10, 1990 this change eliminated the positions of Outage and Modifications Division Manager and Outage Engineer. Based on Shoreham's non-operating, defueled condition, and in accordance with the Agreement between LILCO and the State of New York that LILCO will not operate Shoreham, the positions of Outage and Modifications Division Manager and Outage Engineer are not required because there is no longer a need for planning of scheduled plant outages. The positions of Modification Engineer and Planning and Scheduling Engineer continue to retain their responsibilities as described in the Safety Analysis Report (SAR), but now report to the Operations Division Manager.

These changes, as identified in SNRC-1743, were reviewed and a safety evaluation was prepared pursuant to 10CFR50.59(a)(i) within the broad premise that plant organization and personnel constitute, in part, "a segment of the facility as described in the safety analysis."

Summary

I. No. The changes are administrative in nature and are unrelated to any accident analysis and do not affect the function or operation of any plant system or equipment.

II. No. No physical alterations of plant configuration or changes to setpoints or operating parameters were made. The changes only reflect administrative control changes for outage functional activities.

III. No. The changes are unrelated to any margin of safety.

SEPS 90-007

Description of Change

This change revised the USAR qualification requirement that the Operations Division Manager shall participate in management activities of an operating nuclear plant for two (2) months operation above 20% power and a routine refueling outage (1-2 months) by making them conditional requirements (i.e., the word "shall" was changed to "should"). Shoreham Technical Specification section 6.3.1 and USAR section 3B-1.8 state that the unit staff shall meet or exceed the qualifications of ANSI N18.1-1971. ANSI N18.1-1971 does not require the Operations Division Manager to have participated in the activities as stated above (and as identified in USAR section 13.1.3.2). The change does not delete the qualifications, but does revise the wording to eliminate it as a requirement. This change was made to support the information provided to the NRC via SNRC-1756 dated September 20, 1990.

Summary

I. No. The change is unrelated to any accident analysis and does not affect the function or operation of any plant system or equipment.

II. No. No physical alterations of plant configuration or changes to setpoints or operating parameters were made.

III. No. The change of wording from "shall" to "should" is in accordance with Technical Specification section 6.3.1 and USAR section 3B-1.8. The Operations Division Manager must still meet the qualification requirements of ANSI N18.1-1971. The change is unrelated to any margin of safety.

SEPS 90-008
SPCN 90-0177

Description of Change

A change (modification) was made to RWCU pump G33-P-019B in order to provide cooling to the pump using RWCU process water in lieu of RBCLCW. Additionally, to support this process, a change (modification) was made to the non-safety related portion of the RBCLCW system which feeds the RWCU pump coolers. This overall change is temporary for plant layup purposes. SPCN 90-0177 revised Station Procedure (SP) 23.709.01, "Reactor Water Cleanup System," to allow pump G33-P-019B to be operated with alternate cooling (RWCU) water.

The Shoreham plant is currently defueled with the reactor vessel being maintained at atmospheric conditions with a full water inventory. Technical Specification 3/4.4.4 requires that reactor water purity standards be maintained at all times. The RWCU system performs this task. However, the RBCLCW system has been laid up, and is therefore unavailable to provide cooling to the four cooling loads which are mounted on, and are a part of, the RWCU pump skid. In order to provide an alternate means of cooling and to allow the RWCU pump to operate, the subject modification was engineered. The cooling water is furnished by taking RWCU water from a vent in the pump's discharge, running it through the four cooling loads on the base of the pump, and returning it to the system via a test connection in the pump suction. RBCLCW is positively isolated from the modification by locking closed the isolation valves to and from the RWCU pump skid. Alternate cooling water is connected to the skid via inlet and outlet drain connections already provided, but which normally would be plugged. Cooling of two loads, the bearing cooler and seal cooler, is necessary for maintaining the pump operable. (The other two cooling loads, the pump cover and pedestal, are normally needed only when the reactor is operating.)

This modification allows the RWCU system to be used both for purification of reactor vessel water as well as for water sampling to monitor that purity while the plant is defueled. The intent of this modification is that it only be employed while the reactor is at ambient conditions, and the levels of activity in the coolant remain low.

The RWCU system is designed to process a fraction of reactor coolant flow during normal operation to remove solid and dissolved impurities in order to maintain water purity in accordance with Technical Specification 3/4.4.4. A small portion of the system forms part of the Reactor Coolant Pressure Boundary (RCPB). The cleanup portion of the system, that portion in which the modified pump is located, is <u>not</u> part of the RCPB, per USAR Section 5.1. During normal operation, the RCPB portion of the

SEPS 90-008
SPCN 90-0177
(continued)

system isolates in response to a high differential signal from the RWCU leak detection system, as well as in response to the other signals described in the USAR. All of these signals are unaffected by this modification.

Summary

I. No. This change (modification) does not affect any of the systems associated with previously evaluated accidents such that the probability of occurrence of one of those accidents, or the consequences of one of those accidents will be increased.

II. No. This modification is only used while the plant is in the defueled condition. Therefore, the pressures, temperatures and radiological considerations normally associated with an operating nuclear power plant are absent, and are of no concern for this modification. Also the cleanup portion of the RWCU system, in which the subject pump is located, is not part of the RCPB. The use of process fluid for cooling outside the normal reactor coolant boundary, running it through a non-safety related cooling load, and returning it to the coolant loop, is of concern because water which would normally be considered reactor coolant is being removed from the piping designed to contain it at operating pressures and temperatures, and then being returned to the coolant loop. However, if a leak or rupture in the piping should occur during the defueled condition, it would not pose a safety concern because the levels of activity in the process fluid being used are low. Because the plant is defueled, none of the added piping, or the existing low pressure cooling circuits on the pump base, will be subject to the full reactor operating pressure. In addition, the 150 psig design pressure piping and cooling circuits on the pump base are protected from the maximum expected differential pressure produced by the pump through the use of a restriction orifice and a new relief valve, both conservatively designed to handle maximum anticipated conditions. Finally, the use of a Maintenance Work Request (MWR) requiring dismantling of the modification and decontamination of the affected RBCLW components coupled with the LCO attached to the RBCLCW system, will ensure that the original design configuration will be restored prior to any plant startup or fuel load activity.

SEPS 90-008
SPCN 90-0177
(continued)

III. No. The margin of safety as defined in the bases of any Technical Specification will not be reduced as a result of providing alternate cooling to an RWCU pump while the plant is defueled. This is so because examination of the bases of Technical Specification sections 3/4.4.4 and 3/4.4.5, which discusses water chemistry, which is maintained during layup by the use of RWCU to remove particulates and dissolved impurities via the filter demineralizers, shows that providing alternate cooling to an RWCU pump while the plant is defueled does not affect these bases.

SEPS 90-009

Description of Change

The operating methodology for the RBNVS was revised from a "flow thru" system to a "recirculation" system while SNPS is in a defueled mode. This mode of operation of the Reactor Building HVAC system allowed:

1. The supply portion of the RBNVS to operate in a recirculating mode.

2. The RBSVS cooling portion only to operate.

3. The exhaust portion of the RBNVS to operate partially.

The change was done in order to maintain a 35% relative humidity and a 1/4" - 1" w.g. negative pressure within the Reactor Building during the defueled mode of operation only.

Simultaneous operation of RBNVS and portions of RBSVS necessitates that certain leads to dampers and cooling coil control valves be lifted in order to make the control of these dampers independent of the parent system. In order to operate the RBNVS supply system in a recirculating mode to provide good mixing of the air and to provide heating during the winter, the Reactor Building isolation valves in the RBNVS outside air intake (1T46*AOV-35A/B) will be closed while the access doors are open. To maintain a negative pressure of 1/4" - 1" w.g. within the Reactor Building requires that the isolation valves in the RBNVS exhaust (1T46*AOV-37A/B) be lined up such that one valve is fully open while the second valve in series is cracked open sufficiently to utilize the exhaust fan "head" to create a negative 1/4" - 1" w.g. pressure. The new operating mode with an exhaust air flow rate of approximately 1160 scfm compared to the normal flow rate of 90,000 scfm will affect the operation of 1D11-PNL-029. The isokinetic nozzle in the duct will no longer function as intended since the withdrawal rate of the isokinetic nozzle will be too rapid for the reduced exhaust flow rate. However, virtually all iodines have decayed away, and, with the fuel in the pool, particulates should be retained in the fuel pool. Furthermore, the monitors in this panel do not serve as effluent monitors or fulfill a safety related function. They are intended to detect airborne radioactivity in the Reactor Building. However, there are no detectable gaseous sources at SNPS, either present or anticipated, and the Semi-Annual Effluent Release Report indicates there were no releases from the various building exhaust systems. Any releases from the fuel in the fuel pool would be detected by the safety related refueling level exhaust monitors 1D11-RE-17A/B and 1D11-RE-041 in the station vent. 1D11-RE-029 noble gas monitor would also detect releases, since noble gas monitoring is not affected by deviations from

SEPS 90-009
(continued)

isokinetic sampling, and the noble gas Kr-85 would be the predominant isotope in airborne releases from the fuel. Therefore, the loss of the isokinetic function of the 1D11-PNL-029 nozzle has no safety significance in the current defueled condition. To provide a 35% relative humidity requires that the air circulating within the building be cooled to remove the moisture. The RBSVS cooling coils are therefore required to operate. Modifications to the electrical and instrument air system were necessary to operate the RBNVS and RBSVS as stated above. These changes are fully reversible. The design bases of the RBNVS was changed by this modification.

Summary

I. No. The change in operation of the RBNVS is only acceptable for plant operation in the defueled mode, and must be reversed for any other mode of operation. Therefore, the modification does not affect the functionality of any system intended for safe shutdown of the reactor, or required to mitigate the consequences of an accident, nor does it entail any degradation of any system related to the reactor coolant pressure boundary or primary containment. To operate in any other condition than "defueled" requires that the changes be reversed and hence the functionality of any system intended to mitigate the effect of an accident will not be affected.

II. No. See I above.

III No. See I above.

SEPS 90-010

Description of Change

As described in SNRC-1771 dated November 13, 1990 this change announced a number of Quality Assurance Department organizational changes at the Shoreham Nuclear Power Station. The changes, as identified in SNRC-1771, were reviewed and a safety evaluation was prepared pursuant to 10CFR50.59(a)(i) within the broad premise that the plant organization and personnel constitute, in part, "a segment of the facility as described in the safety analysis."

These changes are as follows:

- The Nuclear Quality Assurance Department (NQAD) has been integrated with the LILCO Corporate Quality Assurance Organization. The Manager, NQAD will report to the Manager of Corporate Quality Assurance for policy matters, personnel administration, coordination of resources and budgetary control. However, the direct operational control and day to day reporting remains, as in the past, to the Vice President - Office of Corporate Services and Vice President - Office of Nuclear. The Manager NQAD continues to have a direct line of communication to the President of the Company for matters concerning nuclear safety.

  The Manager, NQAD will remain as Chairman of the Nuclear Review Board which will continue to function. The NQAD will continue to be located at the SNPS site and has full responsibility for Nuclear QA functions.

- The Safety Engineering and Reliability (SER) Division, since it is no longer required at the Division level, has been removed from the organizational structure. As appropriate this Division's responsibilities have been reassigned or determined to be no longer pertinent. The Reliability Engineering Section is the only functional area to be discontinued. This section previously performed reliability studies of plant systems and components and was a non-regulatory required group which had the objectives of increasing plant operational reliability, availability and maintainability. Under Shoreham's existing non-operational and defueled condition, these objectives serve no useful purpose.

- The Independent Safety Engineering Group (ISEG), which was previously part of the Safety Engineering and Reliability Division, will report directly to the Manager, NQAD, pending NRC approval of License Change Application #8 which requests permission to eliminate the ISEG.

SEPS 90-010
(continued)

Summary

I. No. These changes are administrative in nature and are unrelated to any accident analysis and do not affect the function or operation of any system or component. No program commitments are reduced. The change does not eliminate NQAD, but merely changes the organizational reporting lines.

II. No. No physical alteration of plant configuration or changes to setpoints or operating parameters occurred. The changes do not affect the function or operation of any plant system or equipment. The changes reflect administrative control changes of various functional activities.

III. No. The changes are unrelated to any margin of safety.

SEPS 90-012

Description of Change

This change (ECR H-00727) provided that normally manually closed valves 1G41-04V-0068, 1G41-04V-0069, and 1G41-06V-0070 in the Fuel Pool Cooling and Cleanup System would be locked closed. This change was made in accordance with INPO recommendation (INPO SEN-20, 9/25/87) to prevent inadvertent draining of the Spent Fuel Storage Pool.

Summary

I. No. This change was administrative in nature and does not affect plant operations. The change does not involve any change to the design basis of the plant or any modification to plant structures, systems, or components. The change is unrelated to any accident analysis and does not affect the function or operation of any plant system or equipment.

II. No. See I above.

III. No. See I above.

Nebraska Public Power District

GENERAL OFFICE
P O BOX 499, COLUMBUS, NEBRASKA 68602-0499
TELEPHONE (402) 564-8561
FAX (402) 563-5551

NLS9100189
March 22, 1991

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Subject: Revision to Proposed Change No. 69 to Technical Specifications
Cooper Nuclear Station
NRC Docket No. 50-298, DPR-46

Reference: Letter, L. G. Kuncl to USNRC dated August 31, 1989, "Proposed Change No. 69 to the Cooper Nuclear Technical Specifications, Low-Low Set"

Gentlemen:

In the above reference, the Nebraska Public Power District (the District) submitted a proposed change to the Cooper Nuclear Technical Specifications that would revise the setpoint tolerance on the Low-Low Set Safety/Relief Valve pressure switches. Subsequent NRC staff review of this application generated a comment regarding a bases section in the proposed change. This has been resolved by clarifying the bases section to remove a potential source of confusion. The revised page (Page No. 180) incorporating this change is attached for consideration. The District considers that the no signaficant hazards consideration contained in the reference to still be valid.

If you have any questions, please call.

Sincerely,

G. R. Horn
Nuclear Power Group Manager

GRH/grs:sm
Attachment

cc: Regional Administrator
USNRC - Region IV
Arlington, Texas

NRC Resident Inspector Office
Cooper Nuclear Station

H. R. Borchert
Department of Health
State of Nebraska

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3.7.A & 4.7.A BASES(cont'd)

The primary containment is normally slightly pressurized during periods of reactor operation. Nitrogen used for inerting could leak out of the containment but air could not leak in to increase oxygen concentration. Once the containment is filled with nitrogen to the required concentration, no monitoring of oxygen concentration is necessary. However, at least twice a week the oxygen concentration will be determined as added assurance.

The 500 gallon conservative limit on the nitrogen storage tank assures that adequate time is available to get the tank refilled assuming normal plant operation. The estimated maximum makeup rate is 1500 SCFD which would require about 160 gallons for a 10 day makeup requirement. The normal leak rate should be about 200 SCFD.

3.7.A.6 & 4.7.A.6 LOW-LOW SET RELIEF FUNCTION

The low-low set relief logic is an automatic safety relief valve (SRV) control system designed to mitigate the postulated thrust load concern of subsequent actuations of SRV's during certain transients (such as inadvertant MSIV closure) and small and intermediate break loss-of-coolant accident (LOCA) events. The setpoints used in Section 3.7.A.6.b are based upon a minimum blowdown range to provide adequate time between valve actuations to allow the SRV discharge line high water leg to clear, coupled with consideration of instrument inaccuracy and the main steam isolation valve isolation setpoint.

The as-found setpoint for NBI-PS-51A, the pressure switch controlling the opening of RV-71D, must be ≤ 1040 psig. The as-found closing setpoint for NBI-PS-51B must be at least 90 psig less than 51A, and must be ≥ 850 psig. The as-found setpoint for NBI-PS-51C, pressure switch controlling the opening of RV-71F must be ≤ 1050 psig. The as-found closing setpoint for NBI-PS-51D must be at least 90 psig below 51C, and must be ≥ 850 psig. This ensures that the analytical upper limit for the opening setpoint (1050 psig), the analytical lower limit on the closing setpoint (850 psig) and the analytical limit on the blowdown range (≥ 90 psig) for the Low-Low Set Relief Function are not exceeded. Although the specified instrument setpoint tolerance is ± 20 psig, an instrument drift of ± 25 psig was used in the analysis to ensure adequate margin in determining the valve opening and closing setpoints. The opening setpoint is set such that, if both the lowest set non-LLS S/RV and the highest set of the two LLS S/RVs drift 25 psig in the worst case directions, the LLS S/RVs will still control subsequent S/RV actuations. Likewise, the closing setpoint is set to ensure the LLS S/RV closing setpoint remains above the MSIV low pressure trip. The 90 psig blowdown provides adequate energy release from the vessel to ensure time for the water leg to clear between subsequent S/RV actuations.

3.7.B & 4.7.C STANDBY GAS TREATMENT SYSTEM AND SECONDARY CONTAINMENT

The secondary containment is designed to minimize any ground level release of radioactive materials which might result from a serious accident. The reactor building provides secondary containment during reactor operation when the drywell is sealed and in service. The reactor building provides primary containment when the reactor is shut down and the drywell is open, as during refueling. Because the secondary containment is an integral part of the complete containment system, secondary containment is required at all times that primary containment is required as well as during refueling. Secondary containment may be broken for short periods of time to allow access to the reactor building roof to perform necessary inspections and maintenance.

The standby gas treatment system is designed to filter and exhaust the reactor building atmosphere to the stack during secondary containment isolation conditions. Both standby gas treatment system fans are designed to automatically start upon containment isolation and to maintain the reactor building pressure to the design negative pressure so that all leakage should be in-leakage. Should the system fail to start, the redundant system is designed to start automatically. Each of the two fans has 100 percent capacity.

Arizona Public Service Company

P.O. BOX 53999 - PHOENIX, ARIZONA 85072-3999

WILLIAM F. CONWAY
EXECUTIVE VICE PRESIDENT
NUCLEAR

161-03837-WFC/EST
March 26, 1991

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Mail Station P1-37
Washington, D. C. 20555

Dear Sir:

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Unit 3
Request for a Temporary Waiver of Compliance for Table 3.4-3 of Limiting Condition for Operation 3.4.8.1
File: 91-056-026

Arizona Public Service (APS) respectfully requests a temporary waiver of compliance from the requirements of Table 3.4-3 of Limiting Condition for Operation 3.4.8.1 during the time the reactor vessel head is fully detensioned in the current Unit 3, Cycle 3 refueling outage, as discussed in a conversation with Mr. J. E. Dyer, NRR, on March 25, 1991. A request for a technical specification amendment for all three Palo Verde Units to clarify the basis and applicability of Limiting Condition for Operation 3.4.8.1 will be submitted by May 17, 1991.

Description of Condition

This waiver is requested to allow a cooldown below 93°F. The current allowable cooldown rate of Table 3.4-3 is 0°F per hour rate which would not allow any cooldown below 93°F. The need to have the reactor coolant system (RCS) below this temperature occurs during the removal of the reactor vessel head and filling of the refueling cavity as well as the time period the reactor vessel is defueled and consequently no method of maintaining minimum RCS temperature is available. Plant operations for refueling require the removal of the reactor vessel head and the transfer of the contents of the refueling water tank into the reactor coolant system (RCS) to fill the refueling cavity. The water in the refueling water tank is approximately 70°F and its transfer will result in a gradual cooldown of the RCS to below the 93°F limit of Table 3.4-3 of Limiting Condition for Operation 3.4.8.1, but above the minimum adjusted reference temperature of 45°F. This requirement was introduced in Amendment 24 to the Unit 3 Technical Specifications which was the result of a reanalysis of the pressure-temperature (P-T) limits using the more restrictive requirements of Regulatory Guide 1.99 Revision 2 as required by Generic Letter 88-11. The reanalysis included more restrictive cooldown tables which limited cooldown below 93°F to maintaining an isothermal condition of 0°F per hour with the RCS capable of being pressurized. Prior to this amendment, RCS cooldown was limited to 10°F per hour below 160°F. The maximum allowable cooldown rates specified in Technical Specification 3.4.8 are

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based upon preventing RCS pressure from exceeding the corresponding normal operation pressure-temperature limit, assuming a concurrent pressurization due to the limiting low temperature overpressurization transient. This is not a consideration when the vessel head is fully detensioned and the RCS cannot be pressurized.

Compensatory Actions

Prior to and during the period of this waiver the RCS will be protected from the possibility of an overpressurization event by ensuring the vessel head is fully detensioned. Fully detensioning the vessel head will prevent the RCS from exceeding its P-T limits due to the limiting low temperature overpressurization transient.

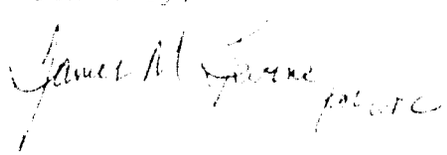
Safety Evaluation

The purpose of the maximum cooldown rates specified in Technical Specification 3/4.4.8 is to prevent the RCS pressure from exceeding the corresponding normal operation P-T limit, assuming a concurrent overpressurization due to the limiting low temperature overpressurization transient. Consequently, the cooldown rate limits are only valid if the RCS is capable of being pressurized. Therefore, with the vessel head detensioned, the technical specification cooldown limits are not applicable. The waiver requested is only for the period of time the vessel head is fully detensioned and thus the RCS cannot be pressurized above the static head of water over the vessel in the refueling pit which is negligible compared to the ASME Code Appendix G, P-T limit.

APS has performed a No Significant Hazards Analysis and Environmental Impact Consideration Determination, Attachment A, and concluded that this waiver of compliance with the compensatory action of fully detensioning the head will have no adverse impact on the continued safe operation of the plant. This wavier request has also been concurred with by the Plant Review Board and ABB Combustion Engineering, the NSSS vendor.

Pursuant to 10 CFR 50.91 (b) (1), and by copy of this letter and attachment, we have notified the Arizona Radiation Regulatory Agency of this request for a waiver of compliance.

Sincerely,



WFC/JST
Attachment

cc: C. M. Trammell (all w/attachment)
J. B. Martin
J. E. Dyer
D. H. Coe
A. C. Gehr
A. H. Gutterman
C. F. Tedford

Attachment A

Basis for No Significant Hazards Consideration

The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92. A proposed waiver of compliance to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with a proposed waiver of compliance would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

A discussion of these standards as they relate to this waiver of compliance request follows:

Standard 1: Involve a significant increase in the probability or consequences of an accident previously evaluated.

The waiving of the requirements of Table 3.4-3 of Technical Specification 3/4.4.8 while the vessel head is fully detensioned does not involve an increase in the probability or consequences of an accident previously evaluated.

The Technical Specification for low temperature overpressure protection is based on the RCS heatup and cooldown rates of Table 3.4-3, as well as the operability of one of the two low temperature overpressure protection shutdown cooling relief valves. Meeting both of these requirements ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 214°F during cooldown and 291°F during heatup. Either one of the two shutdown cooling system (SCS) suction line relief valves provides relieving capacity to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle reactor coolant pump (RCP) with the secondary water temperature of the steam generator less than or equal to 100°F above the RCS cold leg temperatures or (2) the inadvertent safety injection actuation with two high pressure safety injection (HPSI) pumps injecting into a water solid RCS with full charging capacity and with letdown isolated. These events are the most limiting energy and mass addition transients, respectively, when the RCS is at low temperatures.

Waiving Table 3.4-3 requirements when the vessel head is fully detensioned and the RCS cannot be pressurized does not affect either the probability or consequences of the limiting events. This is so, because without the concurrent pressure stress, the thermal stress associated with normal refueling evolutions cannot exceed the P-T limits and therefore no structural integrity issues would exist. Additionally, the adjusted reference temperature for the Unit 3 vessel is 45°F for up to 8 ... Y and thus sufficient margin exists for prevention of brittle fracture.

Standard 2: Create the possibility of a new or different kind of accident from any accident previously evaluated.

The waiving of Table 3.4-3 requirements when the vessel head is fully detensioned does not create the possibility of a new or different kind of accident from any previously evaluated. The change does not affect operation of the plant except to waive cooldown and heatup limits when the reactor vessel cannot be pressurized. During this time period the heatup and cooldown limits are not necessary.

Standard 3: Involve a significant reduction in a margin of safety.

The safety function of the heatup and cooldown limitations are to ensure that the RCS pressure does not exceed the corresponding normal P-T limits, assuming a concurrent pressurization due to the limiting low temperature overpressurization transients described in Standard 1. The waiving of these limitations during the time period the RCS cannot be pressurized does not involve a decrease in the margin of safety. The compensatory action of fully detensioning the head will prevent overpressurization from the limiting low temperature overpressurization transients and results in no decrease in the margin of safety.

Environmental Impact Consideration Determination

The proposed waiver of compliance request does not involve an unreviewed environmental question because operation of PVNGS Unit 3 in accordance with this change, would not:

1. Result in a significant increase in any adverse environmental impact previously evaluated in the Final Environmental Statement (FES) as modified by the staff's testimony to the Atomic Safety and Licensing Board; or

2. Result in a significant change in effluents or power levels; or

3. Result in matters not previously reviewed in the licensing basis for PVNGS which may have a significant environmental impact.

As discussed above, no significant reduction in safety and no new accidents are introduced by this change. This waiver of compliance does not affect effluents, or power levels, and has no environmental impact.

NORTHEAST UTILITIES

NU

General Offices • Selden Street, Berlin, Connecticut

P.O. BOX 270
HARTFORD, CONNECTICUT 06141-0270
(203) 665-5000

March 21, 1991

Docket No. 50-423
B13772
Re: 10CFR50.90

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Gentlemen:

Millstone Nuclear Power Station, Unit No. 3
License Amendment Request--Change in Control of
Public Service Company of New Hampshire's Ownership Interest

Pursuant to 10CFR50.90, Northeast Nuclear Energy Company (NNECO) hereby proposes to amend its facility operating license, NPF-49, to reflect the fact that Public Service Company of New Hampshire (PSNH) will be a wholly owned subsidiary of Northeast Utilities (NU) as of the merger date. Specifically, a new licensee is not being added to the operating license. Rather, NNECO requests that the operating license be amended as shown in Attachment 1 to reflect NU's ownership and control of PSNH as of the merger date.

Millstone Unit No. 3 is a nuclear-powered electric generating facility which has been constructed and is being operated by NNECO on behalf of 14 co-owners (a group of investor-owned and municipal utilities). The NRC operating license, NPF-49, includes a list of co-owners. PSNH, one of the co-owners, is an electric public utility organized and operating under the laws of the State of New Hampshire. PSNH has a 2.8475 percent ownership interest and entitlement in Millstone Unit No. 3.

On January 28, 1988, PSNH filed a voluntary petition with the United States Bankruptcy Court, District of New Hampshire (the "Bankruptcy Court") for protection under Chapter 11 of the Bankruptcy Code and has since operated its business as a debtor-in-possession under the protection of the Bankruptcy Code. After prolonged proceedings, the Bankruptcy Court on December 28, 1989, approved the Third Amended Plan of Reorganization (the "Joint Plan") proposed by Northeast Utilities Service Company and other parties. On April 20, 1990, the Bankruptcy Court confirmed the Joint Plan and ordered its implementation.

Under the terms of the Joint Plan, NU will ultimately acquire all of PSNH's business and assets in exchange for cash and/or securities: PSNH's ownership interest in Unit No. 1 of Seabrook Station ("Seabrook") will be transferred to

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a new wholly owned subsidiary of NU.[1] PSNH's management and operation responsibilities with respect to Seabrook will be transferred to another wholly owned subsidiary of NU;[2] all of PSNH's other assets, including its interest in Millstone Unit No. 3, will remain in the reorganized PSNH which will be merged with yet another subsidiary of NU. Reorganized PSNH will be the surviving entity in such a merger. The existing creditors and equity security holders of PSNH will receive cash and/or securities for their claims or interests. Elements of the Joint Plan are subject to regulatory approval by numerous state and federal agencies. After all regulatory approvals have been received, the Joint Plan contemplates that a formal merger will be consummated. (The date that such merger occurs is referred to herein as the "Merger Date.")

The Millstone Unit No. 3 interest is not technically being transferred as a result of the merger because PSNH, the surviving entity in the merger, will continue to hold this interest after the Merger Date. However, PSNH will become a wholly owned subsidiary of NU after the merger, and thus the Millstone Unit No. 3 interest will undergo a change in control.

On January 23, 1991, pursuant to 10CFR50.80, NNECO requested the consent of the NRC to the acquisition of control by NU of the 2.8475 percent undivided ownership interest of PSNH in Millstone Unit No. 3.[3] In that request, NNECO provided necessary information to ascertain that the proposed action:

1. Will not reduce the funds available to NNECO to carry out activities under its operating license.

2. Will not adversely affect the management of NNECO's operations.

3. Will not result in NNECO being owned, controlled, or dominated by an alien, foreign corporation, or foreign government.

Safety Assessment

The proposed change to the operating license does not involve a change in the safety analysis, the technical specifications, or day-to-day operation of

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(1) T. C. Feigenbaum letter to the U.S. Nuclear Regulatory Commission, "Application to Amend Facility Operating License No. NPF-86 to authorize North Atlantic Energy Corporation as a Licensee to Acquire and Possess the Public Service Company of New Hampshire Ownership Interest in Seabrook," dated November 13, 1990.

(2) T. C. Feigenbaum letter to the U.S. Nuclear Regulatory Commission, "Application to Amend Facility Operating License No. NPF-86 to authorize North Atlantic Energy Service Company to Act as Managing Agent for Seabrook Station, Unit No. 1," dated November 13, 1990.

(3) E. J. Mroczka letter to the U.S. Nuclear Regulatory Commission, "Change in Control of Public Service Company of New Hampshire's Ownership Interest," dated January 23, 1991.

Millstone Unit No. 3. Therefore, the proposed amendment will not affect the safe operation of Millstone Unit No. 3 and is acceptable.

Significant Hazards Consideration

In accordance with 10CFR50.92, NNECO has reviewed the proposed license amendment and concluded it does not involve a significant hazards consideration. The basis for this conclusion is that the three criteria of 10CFR50.92(c) are not compromised. The proposed amendment does not involve a significant hazards consideration because it would not:

1. Involve a significant increase in the probability or consequences of an accident previously analyzed.

   As a result of the proposed license amendment, there will be no physical change to the Millstone Unit No. 3 facility and all limiting conditions for operation, limiting safety system settings, and safety limits specified in the technical specifications will remain unchanged. Also, the Millstone Unit No. 3 Quality Assurance Program and the Emergency Plan, Security Plan, and Operator Training and Requalification Program will be unaffected. In addition, there will be no changes to the operating organization or personnel as a result of the transaction described herein.

2. Create the possibility of a new or different kind of accident from any previously analyzed.

   The proposed amendment will not affect the physical configuration of Millstone Unit No. 3 or the manner in which it will operate. The Millstone Unit No. 3 plant design and design bases will remain the same. The current plant safety analyses will therefore remain complete and accurate in addressing the design basis events and in analyzing plant response and consequences.

   The limiting conditions for operation, limiting safety system settings, and safety limits specified in the technical specifications for Millstone Unit No. 3 are not affected by the proposed license amendment. As such, the plant conditions for which the design basis accident analyses have been performed will remain valid. Therefore, the proposed license amendment cannot create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Involve a significant reduction in a margin of safety.

   Plant safety margins are established through limiting conditions for operation, limiting safety system settings, and safety limits specified in the technical specifications. Since there will be no change in the physical design or operation of the plant, there will be no change to any of these margins. Thus, the proposed license amendment will not involve a significant reduction in a margin of safety.

Moreover, the Commission has provided guidance concerning the application of standards set forth in 10CFR50.92 by providing certain examples (March 6,

1986, 51FR7751) of amendments that are considered not likely to involve a significant hazards consideration. The change proposed herein is most closely enveloped by Example (viii), a change to a license to reflect a minor adjustment in ownership shares among co-owners already shown in the license. As stated earlier, the Millstone Unit No. 3 interest is not technically being transferred as a result of the merger because PSNH, the surviving entity in the merger, will continue to hold this interest after the Merger Date. However, PSNH will become a wholly owned subsidiary of NU after the merger, and thus the Millstone Unit No. 3 interest will undergo a change in control.

Based upon the above and the information in this submittal, there are no significant radiological or nonradiological impacts associated with the proposed amendment, and the proposed amendment will not have a significant affect on the quality of the human environment.

The Millstone Unit No. 3 Nuclear Review Board has reviewed and approved the proposed amendment and concurred with the above determination.

Antitrust Considerations

The merger will not cause any change in the identity of a licensee owning an interest in Millstone Unit No. 3. After the amendment requested herein, the same wholesale customers and the same New Hampshire retail ratepayers will receive the same capacity and energy from Millstone Unit No. 3 that they would have received had the transfer not occurred. Therefore, the transfer of control of PSNH's interest in Millstone Unit No. 3 involves no significant change in the entitlement to the capacity and energy from Millstone Unit No. 3.

PSNH's participation in Millstone Unit No. 3 has already been reviewed as part of the original antitrust review of the project pursuant to Section 105 of the Atomic Energy Act (43 USC 2135). Data with respect to the NU system were also submitted and subjected to review in connection with the licensing of Millstone Unit No. 3 of which NU is, through its subsidiaries, a 65.1715 percent owner. Therefore, since both PSNH and the NU system have undergone an antitrust review in connection with Millstone Unit No. 3, no further review under Appendix L to 10CFR50 is necessary.

Furthermore, to the extent that the acquisition by NU of the entire business of PSNH may raise other questions with respect to antitrust considerations, they have already been reviewed by the Department of Justice and the Federal Trade Commission without any objections being raised. In addition, antitrust considerations of the proposed merger have been extensively addressed in the proceedings conducted by the Federal Energy Regulatory Commission (FERC), in which numerous intervenors actively participated. On December 20, 1990, a FERC Administrative Law Judge issued an initial decision approving NU's proposal to acquire PSNH. The initial decision concludes that the Joint Plan, as proposed with some additional conditions, should be approved as in the public interest. Northeast Utilities Service Company, slip op. at 59 (December 20, 1990) (to be reported at 53 FERC paragraph 963,020).

In addition, on October 5, 1989, an application was filed with the Securities and Exchange Commission (SEC) for approval of those aspects of the Joint Plan

which are subject to its jurisdiction under the Public Utility Holding Company Act of 1935. Again, voluminous pleadings were filed by third parties, including many of the intervenors before the FERC, raising contentions on competitive issues similar to those pursued before FERC. On December 20, 1990, the SEC issued its order approving the aspects of the Joint Plan subject to its jurisdiction, specifically finding the merger would not tend toward the concentration of control of public utility companies of a kind, or to the extent, detrimental to the public interest or the interest of investors or consumers. Northeast Utilities, Public Utility Holding Company Act Release No. 35-25221, slip op. at 40 (December 20, 1990). A petition for rehearing has been filed with the SEC.

Regarding our proposed schedule for this amendment, we request issuance at your earliest convenience, with the amendment to become effective upon issuance.

In accordance with 10CFR50.91(b), we are providing the State of Connecticut with a copy of the proposed amendment.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

E. J. Mroczka
Senior Vice President

cc: T. T. Martin, Region I Administrator
D. H. Jaffe, NRC Project Manager, Millstone Unit Nos. 1 and 3
W. J. Raymond, Senior Resident Inspector, Millstone Unit Nos. 1, 2, and 3

Mr. Kevin McCarthy, Director
Radiation Control Unit
Department of Environmental Protection
Hartford, CT 06106

STATE OF CONNECTICUT)
) ss. Berlin
COUNTY OF HARTFORD)

Then personally appeared before me, E. J. Mroczka, who being duly sworn, did state that he is Senior Vice President of Northeast Nuclear Energy Company, a Licensee herein, that he is authorized to execute and file the foregoing information in the name and on behalf of the Licensee herein, and that the statements contained in said information are true and correct to the best of his knowledge and belief.

Notary Public

Docket No. 50-423
B13772

Attachment 1

Millstone Nuclear Power Station, Unit No. 3

Proposed Change to Operating License

March 1991

NORTHEAST NUCLEAR ENERGY COMPANY, ET AL.[1][2]

DOCKET NO. 50-423

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 3

FACILITY OPERATING LICENSE

License No. NPF-49

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for license filed by Northeast Nuclear Energy Company, as agent and representative of 14 utilities listed below and hereafter referred to as licensees, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I, and all required notifications to other agencies or bodies have been duly made;

   B. Construction of Millstone Nuclear Power Station, Unit No. 3 (the facility) has been substantially completed in conformity with Construction Permit No. CPPR-113 and the application, as amended, the provisions of the Act, and the regulations of the Commission;

   C. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission (except as exempted from compliance, see Section 2.3 below);

   D. There is reasonable assurance: (1) that the activities authorized by this operating license can be conducted without endangering the health and safety of the public, and (2) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I (except as exempted from compliance, see Section 2D below):

---

(1) Northeast Nuclear Energy Company is authorized to act as agent and representative for the following Owners: Central Maine Power Company, Central Vermont Public Service Corporation, Chicopee Municipal Lighting Plant, Connecticut Municipal Electric Energy Cooperative, The Connecticut Light and Power Company, Fitchburg Gas and Electric Light Company, Massachusetts Municipal Wholesale Electric Company, Montaup Electric Company, New England Power Company, Public Service Company of New Hampshire, The United Illuminating Company, The Village of Lyndonville Electric Department, Western Massachusetts Electric Company, and Vermont Electric Generation and Transmission Cooperative, Inc., and has exclusive responsibility and control over the physical construction, operation and maintenance of the facility.

(2) As a result of the Plan of Reorganization confirmed by the U.S. Bankruptcy Court for the District of New Hampshire, Public Service Company of New Hampshire has become a wholly owned subsidiary of Northeast Utilities.

follow-up within thirty days in accordance with the procedures described in 10 CFR 50.73(b), (c), and (e).

G. The licensees shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

H. Except for the provisions of Footnote 2, this license is effective as of the date of issuance. Footnote 2 shall become effective on the date on which Public Service Company of New Hampshire merges with and into a wholly owned subsidiary of Northeast Utilities. This license shall expire at midnight on November 25, 2025.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by H. R. Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Attachments/Appendices
1. Appendix A - Technical Specifications (NUREG-1176)
2. Appendix B - Environmental Protection Plan

Date of Issuance: January 31, 1986

NORTHEAST UTILITIES

General Offices • Selden Street, Berlin, Connecticut

P.O. BOX 270
HARTFORD, CONNECTICUT 06141-0270
(203) 665-5000

March 22, 1991

Docket No. 50-245
B13770
Re: 10CFR50.12
10CFR50, Appendix J

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Gentlemen:

Millstone Nuclear Power Station, Unit No. 1
10CFR50, Appendix J
Request for Schedular Exemption From
Type B and C Test Requirements

Northeast Nuclear Energy Company (NNECO) requested a one-time schedular exemption from the requirements of 10CFR50, Appendix J, Sections III.D.2(a) and III.D.3, on behalf of Millstone Unit No. 1.(1)

This exemption would provide temporary relief from the 2-year schedular requirement associated with Type B and C periodic local leak-rate tests (LLRT). As defined in 10CFR50, Appendix J, Sections III.D.2(a) and III.D.3, Type B and C LLRTs are required to be performed ". . . in no case at intervals greater than 2 years . . ."

During several telephone conversations, the Staff requested additional information regarding the above-mentioned Appendix J schedular exemption request. Specifically, five questions were identified and NNECO's response to these questions is provided below.

NRC Question No. 1:

The Staff requested the last possible date in which Millstone Unit No. 1 could shutdown for the upcoming refueling outage based on fuel burnup predictions with the plant remaining at power operation.

(1) E. J. Mroczka letter to U.S. Nuclear Regulatory Commission, "10CFR50, Appendix J--Request for Schedular Exemption From Type B and C Test Requirements," dated February 5, 1991.

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Response:

Millstone Unit No. 1 is now and should remain operating at approximately 90% power for the remainder of Cycle 13. At this power level, fuel capabilities would support operation until approximately April 24, 1991. However, it should be emphasized that operation of Millstone Unit No. 1 until this date is unlikely because the Cycle 13 outage is scheduled to commence on April 6, 1991, and operation beyond April 10, 1991 would require NRC action to extend the surveillance requirements for snubber inspections.

NRC Question No. 2:

The Staff requested, considering the previous question, how many days will each of the valves requiring local leak rate testing, extend beyond the 2-year requirement.

Response:

As discussed above, Millstone Unit No. 1 is currently scheduled to begin its refueling outage on April 6, 1991. Accordingly, the plant is expected to be shut down prior to the end of the 2-year interval.

A conservative estimation of the maximum number of days which a valve could exceed the 2-year requirement can be determined by utilizing April 8, 1991 (i.e., 2 years from first LLRT performed during the 1989 refueling outage) as the end of the 2-year interval for all valves. Given the uncertainty associated with outage scheduling and to avoid additional action by the Staff, June 30, 1991 has been arbitrarily chosen as the projected Cycle 13 outage completion date in order to establish a reasonable limiting date. Using these assumptions (i.e., April 8, 1991 and June 30, 1991), the 2-year LLRT interval would be exceeded by no more than approximately 83 days. It should be noted that this is a very conservative estimation since not all LLRTs were performed on April 8, 1989, the actual outage end date is expected to be May 23, 1991, and many LLRTs will be completed earlier within the outage.

NRC Question No. 3:

The Staff requested confirmation that valves requiring LLRTs cannot be performed during power operation.

Response:

While there are some nuclear power plants in the industry that have on-line local leak rate testing capability, Millstone Unit No. 1 does not have this capability. Additionally, local leak rate testing with the plant in a shutdown condition is preferable because during power operation the various

valves are performing their intended function which testing would interrupt.

NRC Question No. 4:

The Staff questioned whether past leak rate data would support a conclusion that the valves are expected to be within LLRT criteria.

Response:

Based on available data from past LLRTs at Millstone Unit No. 1, the relatively short period of time in which the 2-year interval would be extended, and improvements which have been made to the Millstone Unit No. 1 LLRT program, NNECO believes that upcoming LLRT results may be consistent with past results and therefore remain within acceptable Appendix J LLRT criteria. However, it should be stressed that it is not possible to predict LLRT results. As requested by the Staff, copies of the penetration leakage calculation data sheets from the 1989 refueling outage are included in Attachment 1. The data sheets include both the "as found" and "as left" leakage.

NRC Question No. 5:

The Staff questioned if an exemption to extend the 2-year Appendix J interval is expected to become a routine request for Millstone Unit No. 1.

Response:

Cycle 14 core exposure capability and anticipated capacity factor result in the next refueling outage being planned for February 1993. Based on this information, NNECO does not currently anticipate that an Appendix J schedular exemption will be required to support the 1993 refueling outage. NNECO intends to closely follow industry and NRC efforts to resolve this generic issue, as licensees extend their fuel cycles. Depending upon currently unknown future fuel cycle extensions for Millstone Unit No. 1, NNECO cannot rule out the possibility of needing to adopt the generic resolution of this issue sometime beyond the 1993 refueling outage.

NNECO trusts that the above information satisfactorily resolves the Staff's questions. We remain available to answer any further questions the Staff may have.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

FOR: E. J. Mroczka
Senior Vice President

BY:
E. A. DeBarba
Vice President

cc: T. T. Martin, Region I Administrator
D. H. Jaffe, NRC Project Manager, Millstone Unit Nos. 1 and 3
W. J. Raymond, Senior Resident Inspector, Millstone Unit Nos. 1, 2, and 3

Docket No. 50-245
B13770

Attachment 1

Millstone Nuclear Power Station, Unit No. 1
10CFR50, Appendix J
Request for Schedular Exemption From
Type B and C Test Requirements

March 1991

PENETRATION LEAKAGE CALCULATION SHEET

1989 [illegible]

SCFH

PENETRATION	"AS FOUND"	"AS LEFT"
DRYWELL HEAD	.13 ± .01 5/9/89	.13 ± .01 5/9/89
X - 1: (Equip. Hatch)	.05 ± .07 5/9/89	.05 ± .07 5/9/89
X - 2: Personnel Air Lock (SP 623.16B) * (TECH SPEC'S EVERY 6 MOS)	*1.56 ± .11	*1.56 ± .11
X - 3: Head Access	.03 ± .01 [illegible]	.03 ± .01 4/20/89
X - 6: CRD Hatch	.03 ± .01 5/6/89	.02 ± .01 5/24/89
X - 8: 1-MS-5, 1-MS-6	.26 ± .11 [illegible]	.03 ± .07 5/12/89
X - 9A: Highest Leakage Rate of either 1-FW-9A or 1-FW-10B	.35 ± .07 4/8/89	.35 ± .07 4/8/89
X - 9B: Highest Leakage Rate of either 1-FW-9B or 1-FW-10B	.03 ± .01 4/9/89	.03 ± .01 4/10/89
X - 10A: 1-IC-2	.03 ± .01 4/9/89	.77 ± .07 5/10/89
X - 11B: 1-IC-3	.03 ± .01 4/9/89	.03 ± .01 4/9/89

PENETRATION LEAKAGE CALCULATION SHEET

PENETRATION	"AS FOUND"	"AS LEFT"
X - 12: 5/19/89 1-SD-2A, 1-SD-2B+ Highest Leakage Rate of either 1-PAS-20 or 1-PAS-17 4/17/89	15.77 ± .74 4/17/89	15.77 ± .74 5/19/89
X - 14: 1-CU-2, 1-CU-2A, 1-CU-5, 1-CU-5	> 76.13 ± .76 4/25/89	8.17 ± .11 5/15/89 3.42 ± .11 5/15/89
X - 15: 1-CU-2B	3.72 ± .11 4/25/89	3.72 ± .11 4/25/89
X - 16A: 1-CS-5A	11.92 ± .74 4/26/89	11.92 ± .74 4/26/89
X - 16B: 1-CS-5B	.33 ± .01 4/29/89	.33 ± .01 4/29/89
X - 17: Highest Leakage Rate of either 1-HS-4 or 1-HS-5	2.13 ± .11 4/27/89	1.01 ± .06 5/14/89
X-24 Highest Leakage Rate of either 1-RC-206 or 1-RC-207	.06 ± .01 5/21/89	.06 ± .01 5/21/89 C H.
X-23 Highest Leakage Rate of either 1-RC-202 or 1-RC-203	.03 ± .01 5/21/89	.03 ± .01 5/21/89
NA 1-IC-6,7	.21 ± .01 4/27/89	.21 ± .01 4/27/89

PENETRATION LEAKAGE CALCULATION SHEET

PENETRATION	"AS FOUND"	"AS LEFT"
X - 25, X-202D:		
1-AC-7, 1-AC-8, 1-AC-9		
1-AC-10, 1-AC-11, 1-AC-12 +		
Inlet Flange of 1-AC-7 +		
Inlet Flange of 1-AC-11 +		
Highest Leak Rate of either		
1-AC-198 or 1-AC-199 +		
Highest Leak Rate of either		
1-AC-206 or 1-AC-205 +		
Highest Leak Rate of either		
1-AC-196 or 1-AC-197.	>76.13±.76 4/12/89	5.58±.10 5/19/89
X-26, X-205:		
1-AC-4, 1-AC-5, 1-AC-6,		
1-AC-17 + 1-AC-2A, 1-AC-3A +		
1-AC-2B, 1-AC-3B + Inlet Flanges of		
1-AC-5 + 1-AC-6 + 1-AC-3A + 1-AC-3B +		
Highest Leak Rate of either		
1-AC-40 or 1-AC-41 +		
Highest Leak Rate of either		
1-AC-133 or 1-AC-134	15.66±.76 4/13/89	15.35±.22 5/17/89

JUL 10 1987

PENETRATION LEAKAGE CALCULATION SHEET

PENETRATION	"AS FOUND"	"AS LEFT"
X - 35A:	.03 ± .01 4/25/89	.03 ± .01 4/25/89
X - 35B:	.42 ± .01 4/25/89	.42 ± .01 4/25/89
X - 35C:	.21 ± .01 4/25/89	.21 ± .01 4/25/89
X - 35D:	.07 ± .01 4/25/89	.07 ± .01 4/25/89
X - 35E:	.03 ± .01 4/25/89	.03 ± .01 4/25/89
X - 36:		
Highest Leak Rate of		
1-AC-139 or 1-AC-140	.37 ± .01 4/10/89	.37 ± .01 4/10/89
X - 37A, B		
X - 38A, B:		
Highest Leak Rate of either		
1-SDV-1S or 1-SDV-2S +		
Highest Leak Rate of either		
1-SDV-3S or 1-SDV-4S	.17 ± .01 4/19/89	.17 ± .01 4/19/89

PENETRATION LEAKAGE CALCULATION SHEET

PENETRATION	"AS FOUND"	"AS LEFT"
X - 37C, D		
X - 38C, D:		
Highest Leak Rate of either		
1-SDV-1N or 1-SDV-2N +		
Highest Leak Rate of either		
1-SDV-3N or 1-SDV-4N	.71 ± .07 4/11/89	.71 ± .07 4/11/
X - 39A:		
1-LP-15A	.27 ± .01 [illegible]	1.36 ± .07 [illegible]
X - 39B:		
1-LP-15B	.03 ± .01 [illegible]	6.41 ± .11 5/31/89
X - 47:		
Highest Leak Rate of either		
1-RR-37 or 1-RR-36	.77 ± .07 4/22/89	.77 ± .07 4/22/
X - 42:		
1-SL-7	6.99 ± .11 4/9/89	6.99 ± .11 4/9/
X - 43:		
1-LP-10A	.28 ± .01 4/10/89	.28 ± .01 4/10/8

PENETRATION LEAKAGE CALCULATION SHEET

PENETRATION	"AS FOUND"	"AS LEFT"
X - 45:		
1-LP-10B, 1-SD-5	4.01 ± .11 5/19/89	4.01 ± .11 5/19/89
X - 101A:	.03 ± .01 4/17/89	.03 ± .01 4/17/89
X - 100D:	.11 ± .01 4/14/89	.11 ± .01 4/14/89
X - 103:	.03 ± .01 4/13/89	.03 ± .01 4/13/89
X - 105D:	.03 ± .01 4/13/89	.03 ± .01 4/13/89
X - 101C:	.07 ± .01 4/13/89	.07 ± .01 4/13/89
X - 104E:	.11 ± .01 4/14/89	.11 ± .01 4/14/89
X - 100E:	3.40 ± .07 4/14/89	3.40 ± .07 4/14/89
X - 104D:	.03 ± .01 4/14/89	.03 ± .01 4/14/89
X - 102:	.04 ± .01 4/14/89	.04 ± .01 4/14/89
X - 105B:	.03 ± .01 4/17/89	.03 ± .01 4/17/89

PENETRATION LEAKAGE CALCULATION SHEET

PENETRATION	"AS FOUND"	"AS LEFT"
X - 108A:	.06 ± .01 4/12/89	.06 ± .01 4/12/[illegible]
X - 104A:	.03 ± .01 4/12/89	.03 ± .01 4/12/[illegible]
X - 100A:	.09 ± .01 4/12/89	.09 ± .01 4/12/[illegible]
X - 106A:	.17 ± .01 4/12/89	.17 ± .01 4/12/[illegible]
X - 100B:	.09 ± .01 4/12/89	.09 ± .01 4/12/[illegible]
X - 104B:	.04 ± .01 4/12/89	.04 ± .01 4/12/[illegible]
X - 104C:	.33 ± .01 4/14/89	.33 ± .01 4/14/89
X - 200A:		
Manhole + (Elec. Penetration if installed)	.03 ± .01 5/4/89 ~~NA~~ 5-23-89 [illegible]	.03 ± .01 5/4/89 ~~NA~~ 5-24-89 [illegible]
X - 200B:		
Manhole + (Elec. Penetration if installed)	.03 ± .01 5/4/89 NA 5-25-89 [illegible]	.03 ± .01 5/4/89 NA 5-25-89 [illegible]

PENETRATION LEAKAGE CALCULATION SHEET

PENETRATION	"AS FOUND"	"AS LEFT"
X - 211A: 1-LP-14A + Highest Leak Rate of either 1-PAS-25 or 1-PAS-24	.57 ± .01 4/10/89	.57 ± .01 4/10/89
X - 211B: 1-LP-14B	.04 ± .01 4/10/89	.04 ± .01 4/10/89
X - 213A: TORUS DRAIN #1	.03 ± .01 4/27/89	.03 ± .01 4/27/89
X - 213B: TORUS DRAIN #2	.03 ± .01 4/27/89	.03 ± .01 4/27/89
X - 202A: Vacuum Breaker "A"	.03 ± .01 4/11/89	.03 ± .01 4/11/89
X - 202B: Vacuum Breaker "B"	.03 ± .01 4/11/89	.03 ± .01 4/11/89
X - 202C: Vacuum Breaker "C"	.03 ± .01 4/11/89	.02 ± .01 5/10/

PENETRATION LEAKAGE CALCULATION SHEET

PENETRATION	"AS FOUND"	"AS LEFT"
X - 202D:	.03 + .01	.01 + .01
Vacuum Breaker "D"		
X - 202E:	.03 + .01	.03 + .01
Vacuum Breaker "E"		
X - 202F:	.03 + .01	.01 + .01
Vacuum Breaker "F"		
X - 202G:		
Highest Leak Rate of either		
1-AC-194 or 1-AC-195 +		
Vacuum Breakers "G" and "I"	.09 + .01	.01 + .01
X - 202H:		
Vacuum Breakers "F" and "J"	.01 + .01	.01 + .01
Gibs #1	10.50 + .11	10.01 + .11
Gibs #2	.03 + .01	.03 + .01
Gibs #3	63.01 + .74	.01 + .01

JUL 16 1987

PENETRATION LEAKAGE CALCULATION SHEET

PENETRATION	"AS FOUND"	"AS LEFT"
Gibs #4	2.3 ± .07	2.3 ± .07
Gibs #5	.11 ± .07	.11 ± .07
Gibs #6	.03 ± .01	.03 ± .01
Gibs #7	.03 ± .01	.03 ± .01
Gibs #8	2.24 ± .07	2.24 ± .07
Total Containment Penetration Leakage	>315.31 ± 1.35 ~~>315.31 ± 1.88~~ 5-17-89 MLL	N/A
Total Containment Penetration Leakage	N/A	125.63 ± 1.15

acceptance criteria is .6 L_a (L_a for Millstone Unit No. 1 is 500.5 SCFH)

[redacted]

MIN PATH
"AS FOUND"
281.03 SCFH

JUL 18 1989

DCR - NRC - COPY 223

March 19, 1991

TO: Manual Holders

FROM: Quad-Cities Station Procedures Coordinator

Please find attached Page 6 of QEP 300-1, Rev. 5. Please insert this page in your manual, which was inadvertently missed for distribution.

Please sign and date below, and return this sheet to: Office Supervisor
Procedures Clerk
Quad-Cities Station.

_____ _____

Signature Date

Alice Beveroth

Alice Beveroth

Procedures

AAB/djb

2/0039j

9103290026 910319
PDR ADOCK 05000254
P PDR

Phone numbers are provided on the State Agency Update Checklist, QEP 300-S6, or the ERF Telephone Directory.

(d) In addition to the verbal transmission above, this information should also be telecopied to the agencies listed above to ensure accuracy if a telecopier is available.

(e) If the ability to telecopy exists, telecopy phone numbers are provided on the State Agency Update Checklist, QEP 300-S6, or the ERF Telephone Directory.

(6) Record and relay inquiries to responsible Directors and the responses to those inquiries.

(7) Act as the primary State/NARS Communicator for the emergency response until the EOF takes command and control, then function in a support role to the EOF Assistant MEO.

(8) Assist the Assistant Station Director in maintaining proper records and logs of GSEP related activities.

d. Technical Communicator (to EOF) - This Communicator reports to the Technical Director. Specific duties include:

(1) Establish and maintain contact with the EOF (once activated) via the Technical Support Party Line (PL-3) or station extension.

(2) Provide the EOF Technical Communicator with plant status information as directed by the Technical Director.

(3) Record and relay inquiries to responsible Directors and the responses to those inquiries.

(4) Assist the Technical Director in maintaining proper records and logs of GSEP related activities.

e. Control Room Communicator (in the TSC) - This Communicator reports to the Operations Director. Specific duties include:

(1) Establish and maintain communications with the Control Room via a dedicated phone line or a station extension.

(2) Communicate and receive plant status information to and from the Control Room, as requested.

(3) Record and relay inquiries to responsible Directors and the responses to those inquiries.

APPROVED

MAR 0 1 1991

Q.C.O.S.R.

Docket No. 70-371
License No. SNM-386

UNC, Inc.
UNC Naval Products Division
ATTN: Mr. Robert Gregg, Director
Technical Services
67 Sandy Desert Road
Uncasville, Connecticut 06382-0981

Gentlemen:

This refers to your submittal to Mr. Jerry Roth of Region I, dated March 18, 1991, which provided the results of your decontamination survey of the concrete lined pit and its access stairway in the area of Unit II. You indicated that the area is now available for the NRC to perform a confirmatory survey.

Our review of your submittal has identified additional information that is needed before final action can be taken. The additional information, specified in the enclosure, should be provided within 7 days of the date of this letter so that the confirmatory survey of the pit area can be performed as scheduled.

Approval of your Decommissioning Plan must be obtained before further confirmatory surveys are performed. The response to our letter of January 17, 1991, is more than 30 days overdue.

If you have any questions regarding this matter, please contact Dr. Sean Soong of my staff at (301) 492-0604.

Sincerely,

Original Signed By

George H. Bidinger, Section Leader
Uranium Fuel Section
Fuel Cycle Safety Branch
Division of Industrial and
Medical Nuclear Safety, NMSS

Enclosure: As stated

Distribution w/encl.

Docket 70-371	**PDR**	**NRC File Center**	IMUF R/F
NMSS R/F	IMSB R/F	Region I	SSoong
VLTharpe	GBidinger	JRoth, RI	

OFC:IMUF:SSoong	IMUF:	RI: Concurred by phone	IMUF:
NAME:SSoong:mh:	VLTharpe:	JRoth:	GHBidinger:
DATE:3/26/91:	3/ /91:	3/ /91:	3/ /91:

OFFICIAL RECORD COPY

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PDR ADOCK 07000371
C PDR

NF15

Request for Additional Information
Application Dated March 18, 1991
UNC, Inc.
Docket No. 70-371

Please provide the following:

1. A description of survey instruments being used and their lower limits of detection.

2. Identification of Radioactive standard sources used for instrument calibration.

3. A description of survey techniques used to generate an average for each grid block.

4. Gamma exposure rate at 1 meter above the surface in each grid in the pit and stairway areas.

5. The reported survey results were obtained for the concrete lined pit area. Is there contamination in or under the liner material? How was this established?

6. A description of the locations of blocks 14EL1 thru 14EL8 and 14PIT9 thru 14PIT23.

7. Controls and procedures to assure that the pit area is not recontaminated.

MAR 2 0 1991

NOTE TO: Don Lanham, IRM, DCB

FROM: Ginny Tharpe, NMSS, IMSB

SUBJECT: RELEASE OF DOCUMENT TO PUBLIC DOCUMENT ROOM

By letter dated January 30, 1991 (copy enclosed), Nuclear Fuel Services (NFS) provided Drawing No. RMG-966D which was marked, "...shall not be used or disclosed..." Per telecon between Scott Pennington, NMSS, and an NFS representative, this drawing can be reproduced and placed in the Public Document Room.

Original Signed By

Ginny Tharpe, NMSS/IMSB

Distribution: w/o encl.
VTharpe SPennington GBidinger NMSS R/F IMUF R/F IMSB R/F
Docket 70-143 NRC File Center

OFC:IMUF:	IMUF:
NAME:VLTharpe:	WSPennington:
DATE:03/21/91:	03/20/91:

OFFICIAL RECORD COPY

1/0

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NF15

71-9132

Department of Energy
Washington, DC 20585

MAR 22 1991

Mr. Charles E. MacDonald
Chief, Transportation Branch
Division of Safeguards and Transportation, NMSS
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. MacDonald:

In reference to our letter of January 25, 1991, requesting a two-year extension of the expiration date for NRC Certificate of Compliance 9132 for the T-3 spent fuel package, the U.S. Department of Energy hereby changes its request to a five-year renewal of the certificate.

If you have any questions on this matter, please contact Steve Primeau of my staff at 353-4210.

Sincerely,

Jacques Read
Acting Director
Division of Quality Verification
and Transportation Safety

cc:
A. Colburn, RL
G. Field, WHC

9103270065 910322
PDR ADOCK 07109132
C PDR

UNCLASSIFIED
FACSIMILE
MESSAGE

U.S. DEPARTMENT OF ENERGY
OFFICE OF QUALITY AND
SAFETY PROGRAMS, EH - 32
WASHINGTON, D.C. 20545

4-E-4

Date: 3/22/90

TO: Bernard White 20455

FROM: Steve Primeau

NO. OF PAGES TO FOLLOW: 1

COMMENTS:

4-E-4

BAKER TESTING SERVICES, INC.
NONDESTRUCTIVE EXAMINATION
METALLURGY AND FAILURE ANALYSIS

98 Reservoir Park Drive
Rockland, MA 02370

(617) 871-4458

Mr. Charles E. MacDonald, Chief
Transportation Certification Branch
Division of Fuel Cycle and Material Safety
Office of Nuclear Material Safety and Safeguards
United States Nuclear Regulatory Commission
Washington, DC 20555

3-22-1991

Dear Mr. MacDonald,

In accordance with 10 CFR 71.12 (c).(3), Baker testing Services, Inc. NRC License Number # 20-19067-01 requests to be registered as a user of Tech/Ops, Inc. Model 660 , Package Indentification Number USA/9033/B(u) Type B (U), under terms of Certificate of Compliance Number 9033 issued to Tech/Ops, Inc., Burlington, Ma.

Sincerely,

Donald Palmer RSO

1/0

9103-90070 910322
PDR ADOCK 07109033
C PDR

N701

NONDESTRUCTIVE EXAMINATION
METALLURGY AND FAILURE ANALYSIS

99 Reservoir Park Drive
Rockland, MA 02370

617 871-4458

4-E-4

FAX TELECOMMUNICATION

DATE: 3-22-91 TIME: 3:15

THIS TELECOPY CONSISTS OF 1 PAGE, (INCLUDING THIS SHEET), AND

IS BEING SENT TO: LEN GORDAN

FROM: DONALD PALMER

COMMENTS: REQUEST FOR RECOMMENDATIONS ON

TELEPHONE NUMBER: (617)871-4458

FAX NUMBER: (617)871-0123

4-E-4

Elouise

71-6272
71-9070
71-9218

SCIENTIFIC ECOLOGY GROUP INC.

March 20, 1991

Mr. Chuck MacDonald
Transportation Branch
Division of Safeguards and Transportation, NMSS
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. MacDonald:

Please register the Scientific Ecology Group (SEG) in accordance with 10 CFR 71.12 as a registered user of the following packages:

NRC Certificate No.	Model
6400 B( )F	Super Tiger
6272 B( )	Poly Panther
9070 B( )F	N-55
9218 B(U)F	TRUPact II

We will operate these packages in accordance with the SEG QA program SEG/QA-100 (NRC approval 0496) and the pertinent certification, certification drawings, and procedures.

Please advise me if you have questions.

Sincerely,

Bryan Roy
Engineering Manager

BAR/mdf
ltr-91-056

cc: Jim Mooney
Walt Hipsher
Tim Ramsey
Roger Betow
Jay Pride

1/0

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PDR ADOCK 07106272
C PDR

P.O. Box 9530 1560 Bear Creek Rd Oak Ridge, Tennessee 37831-9530 (615) 481-0222

NT01

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

MAR 2 2 1991

SGTB:CTH
71-0122

J. L. Shepherd & Associates
ATTN: Mr. J. L. Shepherd
President/Owner
1010 Arroyo Avenue
San Fernando, California 91340

SUBJECT: INSPECTION REPORT NO. 710122/90-05, NOTICE OF VIOLATION, AND NOTICE OF NONCONFORMANCE

Dear Mr. Shepherd:

This refers to the inspection conducted by Mr. L. L. Gordon and Mrs. C. Haney of this office on January 28-30, 1991. The inspection consisted of an examination of procedures and representative records, interviews with personnel, and observations by the inspectors of activities conducted at your San Leandro, California facility. At the conclusion of the inspection, the findings were discussed with members of your staff.

Notice of Violation

Based on the results of this inspection, certain of your activities appeared to be in violation of U.S. Nuclear Regulatory Commission (NRC) requirements, as specified in the enclosed Notice of Violation.

This violation has been categorized by severity level in accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," 10 CFR Part 2, Appendix C (1988). This Notice of Violation is sent to you pursuant to the provisions of Section 2.201 of the NRC's "Rules of Practice," Part 2, Title 10, Code of Federal Regulations. You are required to respond to this letter and should follow the instructions specified in the Notice of Violation.

Notice of Nonconformance

During this inspection it was also found that the implementation of your Quality Assurance Program failed to meet certain NRC requirements. The specific findings and references to the pertinent requirements are identified in the enclsosed Notice of Nonconformance and Inspection Report.

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Notice of Nonconformance

Please provide us, within 30 days from the date of this letter, a written response to the enclosed Notice of Nonconformance containing: (1) a description of steps that have been or will be taken to correct these items; (2) a description of steps that have been or will be taken to prevent recurrence; and (3) the dates your corrective actions and preventive measures were or will be completed. We will consider extending the response time for the Notice of Nonconformance if you can show good cause for us to do so.

The responses directed by this letter and the enclosed Notices of Violation and Nonconformance are not subject to the clearance procedures of the Office of Management and Budget as required by the Paper Work Reduction Act of 1980, P. L. No. 96-511.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," Part 2, Title 10, Code of Federal Regulations, a copy of this letter and its enclosures will be placed in the NRC Public Document Room.

Should you have an questions about this inspection, we will be pleased to discuss them with you.

Sincerely,

Original Signed by

Charles E. MacDonald, Chief
Transportation Branch
Division of Safeguards
and Transportation, NMSS

Enclosures:
1. Inspection Report No. 710122/91-05
2. Notice of Violation
3. Notice of Nonconformance

Reviewed by:
E. Kraus, NMSS 02/24/91 G. Bennington, SGTB 02/ /91

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

CONTAINER SUPPLIER INSPECTION PROGRAM

Inspection Report

ORGANIZATION: J. L. Shepherd & Associates
ADDRESS: 1010 Arroyo Avenue
San Fernando, California 91340

CONTACT: Mr. J. L. Shepherd
TITLE: President, Owner
TELEPHONE: 818-898-2361

ACTIVITY: Fabricate, use, and design radioactive material packages

QUALITY ASSURANCE PROGRAM APPROVAL NO.: 0122

INSPECTION REPORT NO.: 710122/91-05

Inspection Location:	Inspection Dates:	Inspection On-site Person-Hrs.:
San Leandro, California	1/28 - 1/30/91	34

INSPECTION BASES AND SCOPE:

A. BASES: 10 CFR Part 71, 10 CFR Part 21, Certificate of Compliance No. 5984, and 6280.

B. SCOPE: To determine whether the organization has established, documented and executed procedures which fulfill the commitments made in the organization's NRC-approved quality assurance program.

To determine whether packages are fabricated in accordance with the design approved by the Commission.

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FINDINGS:

A violation with the requirement of 10 CFR 21.21(a) was identified.

Nonconformances with the requirements of 10 CFR 71.105(d), 71.107(c), 71.109, 71.111, 71.113, and 71.135 were identified.

INSPECTION TEAM LEADER: Len Gordon DATE: 3-7-91
Len L. Gordon
NMSS

INSPECTOR: Catherine T Haney DATE: 3-14-91
Catherine T. Haney
NMSS

REPORT APPROVED BY: Charles E. MacDonald DATE: 3/22/91
Charles E. MacDonald
Chief, Transportation
Branch, NMSS

1. SUMMARY

A U.S. Nuclear Regulatory Commission (NRC) inspection team conducted an announced inspection of J. L. Shepherd and Associates (JLS&A) in San Fernando, California, between January 28-30, 1991. Inspection findings are based on data collected through observation of selected activities, review of implementation procedures and controls, review of selected documents and records, and interviews with personnel. The inspection team concluded that the implementation of the quality assurance (QA) program was satisfactory, in general. However, it identified a violation of Reporting of Defects and Noncompliance (10 CFR 21.21(a)) and items of nonconformance with some aspects of QA Management (10 CFR 71.105(d), 71.113, 71.135, and 71.137); Design Control (10 CFR 71.107(c), and Maintenance Control (10 CFR 71.111).

The team discussed tentative findings with the organization's management at the exit meeting.

2. DETAILS

The team inspected packaging design, fabrication, maintenance, and facility management activities at the JLS&A facility in San Fernando, California. The objectives of the inspection were to determine how well the JLS&A QA Program complied with commitments made to NRC, to review the implementation of the QA Program, and to verify whether the products were fabricated and maintained in compliance with NRC requirements. The team evaluated QA activities from the perspective of four functional elements: (1) Management Controls; (2) Design Controls; (3) Fabrication Controls; and (4) Maintenance Controls. The inspection focused on the QA activities of those packaging models that are in commercial use; specifically Model 5984 (Certificate of Compliance No. 5984), Model A-0109 Irradiator in A-0117 Overpack (Certificate of Compliance No. 6280) and the Department of Transportation (DOT) SPEC 20 WC-5 and 20 WC-6 overpacks.

This inspection was based on the guidance provided in the NRC Division of Safeguards and Transportation Temporary Instruction "Transportation Package Suppliers Inspection," the Technical Report "Packaging Supplier Inspection Guide," and the Package Supplier Inspection Tree (see Figure 1). The inspection tree was used as a visual aid during the exit meeting.

2.1 Persons Contacted

An entrance briefing, presenting the scope of the inspection and the use of the Management Oversight Risk Tree (MORT) in evaluating the organization's performance, was held on January 28, 1991. An exit briefing, presenting the preliminary findings of the inspection, was held on January 30, 1991. The following individuals were present at the entrance and/or exit meetings.

*J. L. Shepherd, Owner, President
*M. F. Shepherd, Vice-President
*J. S. Shepherd, Director, Business Development
**J. Fuzzell, Director, Radiological Services
**D. E. Bunn, Senior Health Physicist, State of California
***D. Shepherd, Vice-President

2.2 Management Controls

The inspection of this element focused on two primary subjects: the extent to which JLS&A has implemented "Quality Assurance Approval for Radioactive Material Packages," No. 0122, and the independence and proficiency of the QA personnel responsible for fulfilling the QA commitment. The inspectors evaluated program effectiveness by (1) examining QA documents, applicable administrative procedures, completed manufacturing documents, and contractor inspection reports; and (2) interviewing staff members.

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*Present at entrance and exit briefings
**Present at entrance briefing, only
***Present at exit briefing, only

The JLS&A primary product line is the manufacture of calibrators and irradiators, which are designed to meet specifications for DOT Type 7A containers, for its own use and the use of its customers. The use of the two NRC-certified containers, Model 5984 and Model A-0117, are essentially designed to facilitate transport of JLS&A DOT 7A packages and Model A-0109 irradiators, respectively. Because of the specialized nature of JLS&A business, major effort is applied to the radiological safety program and a somewhat more limited effort is directed to ensure that the QA requirements of 10 CFR Part 71 are met. However, management indicated to the inspection team that management closely scrutinizes and oversees functions and activities necessary to ensure that the quality objectives have been met and that the QA Program provides prompt resolution to unsatisfactory work.

The JLS&A QA Program Plan, "QA-RM-001-A," Revision 3, dated October 10, 1990, approved by NRC, describes a comprehensive commitment to meet all the QA criteria identified in 10 CFR Part 71, Subpart H, as supplemented by the guidance of NRC Regulatory Guide 7.10. However, the implementation of the QA program by written procedures was found deficient in a number of areas, including QA management oversight, document control, records management, and the audit program as described below.

A nonconformance was identified with regard to 10 CFR 71.113, "Document control," which requires that measures be established to control the issuance of documents such as instructions, procedures, and drawings, including changes, which prescribe all activities affecting quality. The NRC inspection team reviewed the acceptance tests, operating, and maintenance procedures referenced on the Certificates of Compliance and numerous forms/checksheets, to verify that approved procedures were being used. The procedures had no date of issue or revision number. The lack of this information made it impossible for the inspection team to identify if the procedure was current. This condition could result in JLS&A using outdated or obsolete procedures. In addition there was no indication that management had reviewed or approved procedures or data sheets.

A nonconformance was identified with regard to 71.135, "Quality assurance records," which requires that written records of quality-related activities be maintained for three years after personnel have last engaged in the activity for which the QA Program had been issued. It also requires that a record retention program be established by instruction or procedure. This should also take into account the requirements of 10 CFR 71.91 to ensure that all required records will be retained. The following nonconformances were identified:

a. The NRC inspectors asked to review the file containing nonconformance and corrective action reports. However, at the time that the inspection was concluded, the nonconformance file still had not been located.

b. The review of the records filing system indicated that records were not controlled, and that responsibility for controlling records is not defined and not assigned to any JLS&A staff member. The JLS&A "Statement of Verification" related to QA Records indicate that files on reusable shipping containers may be maintained for a 2 year period.

Examination of the audit files indicated that audits of transportation and packaging activities were performed on an annual basis, the last being on October 8, 1990. Audits were conducted by using a checklist, "Internal

Audit," and included calibration of measuring instruments, inspection and test status, purchasing, electrical department, and manufacturing control.

Nonconformances were identified with regard to 10 CFR 71.137, "Audits," which requires that a comprehensive system of planned and periodic audits be conducted to verify compliance with all aspects of the QA program. These audits must be performed in accordance with written procedures or checklists by trained personnel. In addition, the JLS&A, "Quality Assurance Program Plan," Revision 3, Annex 1, 1.18 and Annex 2, 2.18, states that pre- and post-audit conferences are to be held, corrective action reports written, and timely response and followup action taken, to ensure that an appropriate corrective action was accomplished. A review of the Audit program resulted in the following nonconformances:

a. The audit file showed no evidence of a planned audit program that identifies the scope of the audit, provisions for pre- and post-audit conferences with management on the activities under audit, scheduled intervals for auditing of each packaging and transportation activity, or followup of corrective actions taken as a result of the identified program deficiencies.

b. Documentation supporting verification of qualifications of the designated lead auditors was unavailable. JLS&A committed, in its application for a QA Program, to following the requirements, as specified in Supplement 2S-3 to ANSI/ASME NQA-1.

JLS&A performs external/vendor audits as a means of placing a vendor or subcontractor on the approved vendor list (AVL). The "Quality Assurance Program Plan," dated October 10, 1990, states in Sections 1.18 and 2.18 "Audits," Paragraph 2, "External surveys or audits of major contractor's or supplier's facilities are performed at the time of initial contract/purchase, per Section 1.7 & 2.7, 3, and contractors and suppliers will be re-evaluated if specifications change, in the case of non-conformances or corrective actions or after a license or approval is renewed." In practice, JLS&A qualifies vendors by accepting the results of audits done by other organizations or by its past experience with the vendor. Typically, a vendor is not removed from the AVL unless JLS&A notes a deterioration in the product goods or services supplied by the vendor. The team noted that the vendors used to fabricate the SPEC 20 WC-6 overpack, Serial Number 22281, were on the AVL. Documentation of audits done by other organizations were available to support placing them on the AVL.

A nonconformance was identified with 10 CFR 71.105(d), "Quality assurance program," which requires that training be provided to employees to ensure that proficiency is achieved and maintained. The team reviewed the training file and confirmed that personnel are certified in performing radiological safety activities by examination. However, there was no evidence of any orientation related to 10 CFR Parts 21 or 71, Subpart H requirements.

A violation was identified with regard to 10 CFR 21.21(a), "Notification of failure to comply or existence of a defect," which requires that procedures be established to: (1) provide for evaluating deviations or informing the licensee or purchaser of the deviation; and (2) ensure that a responsible officer is informed if the component: (a) fails to comply with the Atomic Energy Act of 1954, as amended, or any rule, regulation, order, or license of the Commission; or (b) contains a defect. Contrary to this requirement, a formal procedure was not available.

2.3 Design Control

The design-modification process was reviewed to verify that adequate controls had been developed and implemented to ensure compliance with the Safety Analysis Report (SAR) and QA Program requirements. The team reviewed the process for initiating major and minor design modifications, in order to verify that adequate controls had been developed and implemented. These controls should be used to determine if prospective design changes have an impact on the commitments made in the SAR or conditions listed on the Certificate of Compliance.

A nonconformance was identified with regard to 10 CFR 71.107(c), "Package design control," which requires that design changes, including field changes, be subject to the design control measures commensurate with those applied to the original design. JLS&A management was not aware that a change in the Model 109 irradiator may require an amendment to the corresponding NRC Certificate of Compliance. JLS&A did not have a formal procedure for ensuring that design changes be reviewed commensurate with the original design.

2.4 Fabrication Control

The team inspected fabrication records to verify that all phases of the fabrication process were properly controlled and implemented. Fabrication records were reviewed to verify that that the prescribed inspection/testing was accomplished.

JLS&A inventory, according to the "Overpack List," dated January 19, 1991, includes two NRC certified packagings (one Model 5984 and one Model A-0117 overpack) and eighteen DOT Specification overpacks (two SPEC 20 WC-3, eight SPEC 20-WC-5, and eight SPEC 20 WC-6). The NRC inspection team reviewed fabrication files for the NRC certified packagings and the DOT 20 WC-5 and WC-6 overpacks.

2.4.1 NRC Certified Packagings

JLS&A fabricated one Model 5984 overpack between 1968 and 1969 and one Model A-0117 overpack during 1979. Both units were fabricated before the requirements for an NRC-approved QA Program. The overpacks have been repaired as a result of maintenance inspections, and no modifications have been made to the overpacks since their initial fabrication.

Fabrication records for each unit contained QA documents such as purchase orders for the materials and/or purchase orders for the fabrication of the overpack, material certifications, and QA data checklists. A notation was found, in the files, that stated that on September 19, 1990, an internal audit was performed of records contained in the files, and applicable QA documentation, which may have been missing from the files, was added.

2.4.2 DOT Specification Packagings

JLS&A fabricates, uses, and maintains DOT 20 WC-5 and 20 WC-6 packagings, as described in 49 CFR 178.194. JLS&A certifies, to users of the overpacks, that the overpack is constructed in accordance with 49 CFR 178.194 and is maintained in accordance with the JLS&A NRC-approved QA Program. The NRC inspection team

reviewed fabrication files for one DOT 20-WC-5 and one DOT 20 WC-6 packaging. It was noted that the files had been reviewed and updated during the JLS&A internal audit on September 19, 1990.

Records for overpacks fabricated in 1981, 1987, and 1990 were reviewed. The records were selected as samples to identify variations in record keeping over the years, the methods of fabrication, and the completeness of the files. All three files contained similar types of documents and checksheets. However, the inspection team noted that record keeping and the quality of the documentation improved over the years.

The record review showed that JLS&A fabricated parts for some units, wheras, in other cases, subcontractors were used to fabricate either the components of the overpacks or the complete overpack. One such example is the DOT 20 WC-6 Overpack Serial Number 22281, fabricated in 1990. Specifically, West Coast Packing and Crating, operating under the provisions of JLS&A Purchase Order No. 3149, dated June 26, 1990, fabricated the wooden overpack, whereas the steel shell was procured from Burbank Sheet Metal, operating under the provisions of Purchase Order No. 3305, dated August 1, 1990. JLS&A welded the shell and assembled the unit. Other components of the overpack were taken from stock inventory and were identified by purchase orders. A QA check sheet was used for final acceptance of the SPEC 20 WC-6 Overpack. Items such as the inner and outer dimensions, inside and overall height, side wall, and end thickness were measured. J. L. Shepherd performed the acceptance testing.

A nonconformance was identified with regard to 10 CFR 71.109, "Procurement document control," which requires that measures be established to ensure that adequate quality is required in the documents for procurement of material, equipment, and services, whether purchased directly or through contractors or subcontractors. JLS&A Purchase Order 3149 was issued to West Coast Packing and Crating to fabricate DOT 20-WC-6, Serial Number 22281. The specifications on the purchase order were described as follows: "DOT 20 WC-6 Overpack less 12-ga steel as stated in 178.192.2(2). . . ." The proper reference should have been 49 CFR 178.194-2(2). There was no technical review of the specification before issuing the purchase order.

2.5 Maintenance Controls

The inspection team reviewed records on maintenance that is to be performed on the packagings and confirmed that the scheduled maintenance had been performed. The inspectors reviewed maintenance documents for completeness and adequacy of identified maintenance, maintenance controls, and QA Program controls.

2.5.1 Maintenance of NRC Certified Packagings

Maintenance records for the packaging covered by Certificate of Compliance 6280 were reviewed. Certificate of Compliance 5984 was not reviewed, since this overpack has not been used since 1988, and there are no plans to use the unit. JLS&A specifies the maintenance requirements for the overpack in various documents as follows:

Certificate of Compliance 6280, dated February 21, 1990:

#7(a) The package must be maintained in accordance with the Maintenance Program described in the JLS&A submittal dated February 2, 1990.

JLS&A Submittal dated February 2, 1990:

Excerpted from "Procedures for Removal of the Irradiator from the Model A-0117 Overpack, COC 6280" - Item E "A-0117" Overpack Checkout and Maintenance Procedures"

Each time the overpack is received and prior to each shipment, it is thoroughly inspected according to the attached check list...

Application for QA Program, dated March 30, 1983 (in effect until November 27, 1990)

"Quality Assurance Procedures for Design, Fabrication and Use of Shipping Containers and Overpacks Used for the Transportation of Radioactive Materials," Number QA-RM-001-A, Revision 1.

Excerpted from Section XIII. "Handling, Storage, & Shipping":
J. L. Shepherd & Associates performs a final inspection before shipment on all items pertinent to NRC and U.S. DOT specifications and conditions. Items must pass all criteria before shipment is made.

Nonconformances were identified with 10 CFR 71.111,"Instructions, procedures and drawings," which requires that activities affecting quality be documented by instructions, procedures, or drawings. These instructions, procedures, or drawings must be followed and must include quantitative or qualitative acceptance criteria. The nonconformances are as follows:

a. An inconsistency was noted with how the Overpack Check-In List was being used to record the survey results. The inconsistency involved the recording of the status of the locking ring and the removable rods on the Model A-0117 Overpack, Serial Number 22197. No uniform acceptance criteria were available for evaluating the status of the locking ring and the removable rods. A review of the maintenance file for Overpack Serial Number 22197 showed that one inspector typically recorded the condition of the locking ring and removable rods to be "NA" (not applicable), whereas another inspector recorded the condition as "cleaned" or " ". The team was informed that locking rings and removable rods do not apply to this unit.

b. JLS&A, in letter dated February 2, 1990, committed to using a specific check list for performing check-in and shipping surveys on the overpacks. This form was not being used to survey the units. This discrepancy was brought to the attention of management during the inspection. A nonconformance report was generated before the close of the inspection.

c. Check-in or maintenance records were not available for the following shipments:

March 5, 1990	General Electric to JLS&A (JLS&A was shipper of record)
March 8, 1990	JLS&A to General Electric
March 15, 1990	JLS&A to General Electric
March 22, 1990	General Electric to JLS&A

2.5.2 Maintenance on DOT 20 WC-5 and 20 WC-6 Packagings

The maintenance requirements for the DOT WC-5 and 20 WC-6 overpacks are found in the application for the JLS&A QA Program, as follows:

Application for QA Program, dated March 30, 1983 (in effect until November 27, 1990)

"Quality Assurance Procedures for Design, Fabrication, and Use of Shipping Containers and Overpacks Used for the Transportation of Radioactive Materials," Number QA-RM-001-A, Revision 1.

Excerpted from Section XIII, "Handling, Storage, & Shipping":
J. L. Shepherd & Associates performs a final inspection before shipment on all items pertinent to NRC and U.S. DOT specifications and conditions. Items must pass all criteria before shipment is made.

JLS&A must comply with these statements when they are maintaining or using the overpacks at an NRC licensee, i.e., performing the work under reciprocity (10 CFR Part 150).

Nonconformances were identified with 10 CFR 71.111, "Instructions, procedures, and drawings," which requires that activities affecting quality be documented by instructions, procedures, or drawings. These instructions, procedures or drawings must be followed and must include quantitative or qualitative acceptance criteria. The nonconformances are as follow:

a. A preshipment or maintenance record was not available for shipments involving the use of DOT 20 WC-6 Overpack Serial Number 22280. The shipments are as follow:

October 22, 1990	Sandoz to JLS&A
November 14-21, 1990	Hutchinson to JLS&A

b. There were no objective criteria for performing the inspections. Two Check-In Lists, Form QA-005-OPC, were found for the shipment of the DOT 20 WC-6 Overpack Serial Number 22280, on November 21, 1990. Each inspection form was completed by a different inspector, and the results of the inspections did not agree.

3. DOCUMENTS REVIEWED

Certificates of Compliance for Radioactive Materials Packages and Supporting Documentation

Certificate Number 5984, Revision 5
Certificate Number 6280, Revision 5

Quality Assurance Program Approval for Radioactive Material Packages, Number 0122, Revision 3 and 4

Quality Assurance Program Plan, QA-RM-001-A, dated October 10, 1990 and March 30, 1983

Vendor Analysis Report, dated November 30, 1990

JLS&A, California Radioactive Material License, No. 1770-70, Amendment 57

10 CFR 21.6 Posting Requirements

Overpack Check-In List, Form QA-005-OPC

Shipments of Overpack 6280, SN 22197, November 15, 1989 to September 4, 1990
Shipments of Overpack DOT 20WC- , SN 22280, September 25, 1990 to November 21, 1990

Purchase Orders

PO 615-5968, dated July 8, 1970, JLS&A to Morrison Engineering Co.
PO 3305, dated August 1, 1990, JLS&A to Burbank Sheet Metal
PO 3149, dated July 26, 1990, JLS&A to West Coast Packing and Crating
PO 3302, dated August 1, 1990, JLS&A to Tube Service

Fabrication Files

DOT 20 WC-5, SN 22196
DOT 20 WC-6, SN 22281
Model 5984 Overpack, SN
Model A-0117 Overpack, SN

Nonconformance Report, dated January 29, 1991

Certification, dated March 13, 1990

Overpack List, dated January 19, 1991

Instructions and Procedures

"Manufacturing Control", Instructions, Procedures and Drawings, Subpart H of Part 71, Criteria 5 and 6

"Design Control", Instructions, Procedures and Drawings, Subpart H of Part 71, Criterion 3

"Job Descriptions", Criteria 1

Forms

Material Rejection

Internal Audit Checklist

Welder Certification Sheets

Handling, Storage and Shipping, QP 13.0, dated September 9/30, 1990.

Reports

Telephone Systems Co., Northridge, CA.

Includes: Micrometer, Model OD-0-1
Micrometer, Model arvl 0-6
Dial Caliper, Model 0-8

APPENDIX A

J. L. Shepherd and Associates
Docket No. 71-0122

NOTICE OF VIOLATION

During a U.S. Nuclear Regulatory Commission (NRC) inspection conducted between January 28-30, 1991, a violation of NRC requirements was identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," 10 CFR Part 2, Appendix C (1988), the violation is listed below:

1. 10 CFR 21.21(a) requires that procedures be established to: (1) provide for evaluating deviations or informing the licensee or purchaser of the deviation; and (2) ensure that a responsible officer is informed if the component: (a) fails to comply with the Atomic Energy Act of 1954, as amended, or any rule, regulation, order or license of the Commission; or (b) contains a defect.

   Contrary to the above, J. L. Shepherd & Associates did not have a formal procedure that met the requirements of 10 CFR 21.21(a).

   This is a Severity Level IV violation (Supplement VII).

Pursuant to the provisions of 10 CFR 2.201, J. L. Shepherd & Associates is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555 with a copy to the Branch Chief, Transportation Branch, Office of Nuclear Material Safety and Safeguards (NMSS), within 30 days of the date of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation" and should include for each violation: (1) admission or denial of the alleged violation; (2) the reason for the violation, if admitted; (3) the corrective steps that have been taken and the results achieved; (4) the corrective steps that will be taken to avoid further violations; and (5) the date when full compliance will be achieved. If an adequate reply is not received within the time specified in this Notice, an order may be issued to show cause why the Quality Assurance Program Approval should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

FOR THE NUCLEAR REGULATORY COMMISSION

Charles E. MacDonald, Chief
Transportation Branch, NMSS

Dated at Rockville, Maryland
This 22 day of March 1991

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PDR ADOCK 07100122
C PDR

APPENDIX B

J. L. Shepherd & Associates
Docket No. 71-0122

NOTICE OF NONCONFORMANCE

Based on the results of a U.S. Nuclear Regulatory Commission (NRC) inspection conducted between January 28-30, 1991, it appears that certain of your activities were not conducted in accordance with NRC requirements.

1. 10 CFR 71.105(d) requires that training be provided to employees to ensure that proficiency is achieved and maintained.

   Contrary to the above, a training program, specific to the requirements of 10 CFR Part 21 and Part 71, Subpart H, had not been developed.

2. 10 CFR 71.107(c) requires that design changes, including field changes, be subject to the design control measures commensurate with those applied to the original design.

   Contrary to the above, formal design control procedures were not available for ensuring that design changes were reviewed commensurate with the original design.

3. 10 CFR 71.109 requires that measures be established to ensure that adequate quality is required in the documents for procurement of material, equipment, and services, whether purchased directly or through contractors or subcontractors.

   Contrary to the above, Purchase Order 3149, issued to West Coast Packing & Crating, did not undergo a technical review before being issued. The purchase order incorrectly referenced "178.192.2(2)" rather than "49 CFR 178.194-2(2)."

4. 10 CFR 71.111 requires that activities affecting quality be documented by instructions, procedures, or drawings. These instructions, procedures, or drawings must be followed and must include quantitative or qualitative acceptance criteria.

   NRC Certificate of Compliance 6280, dated February 21, 1990, states:

   > Condition 7(a) This package must be maintained in accordance with the Maintenance Program described in the JLS&A submittal dated February 2, 1990.

   J. L. Shepherd & Associates (JLS&A) submittal dated February 2, 1990, states:

   > Each time the overpack is received and prior to each shipment, it is thoroughly inspected according to the attached checklist...

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PDR ADOCK 07100122
C PDR

Application for Quality Assurance (QA) Program, dated March 30, 1983

Section XIII. Handling, Storage, & Shipping
J. L. Shepherd & Associates performs a final inspection before shipment on all items pertinent to NRC and U.S. DOT specifications and conditions. Items must pass all criteria before shipment is made.

Contrary to the above,

a. The Overpack Check-In List was being completed inconsistently for the Model A-0117 Overpack, Serial Number 22197. There were no uniform acceptance criteria for evaluating the status of the locking ring and the removable rods.

b. The checklist specified by JLS&A in its letter of February 2, 1990, was not being used to survey the Model A-0117 Overpack.

c. Check-in or maintenance records were not available for the following Model A-0117 Overpack shipments:

1.	March 5, 1990	General Electric to JLS&A*
2.	March 8, 1990	JLS&A to General Electric
3.	March 15, 1990	JLS&A to General Electric
4.	March 22, 1990	General Electric to JLS&A*

*According to the staff, JLS&A was the shipper of record.

d. A pre-shipment or maintenance record was not available for shipment involving the use of the Department of Transportation, DOT 20 WC-6 Overpack, Serial Number 22280:

1.	October 22, 1990	Sandoz to JLS&A
2.	November 14-21, 1990*	Hutchinson to JLS&A

*Exact date of shipment was not available.

e. There were no objective criteria for performing the pre-shipment or maintenance surveys for the DOT 20 WC-5 and WC-6 Overpacks. Two Check-In Lists, Form QA-005-OPC, were found for the shipment of the DOT 20 WC-6, Serial Number 22280 Overpack on November 21, 1990. The surveys were performed by different inspectors. Each form had different results for same inspection items.

5. 10 CFR 71.113 requires that measures be established to control the issuance of documents such as instructions, procedures, and drawings, including changes, that prescribe all activities affecting quality.

Contrary to the above, acceptance test, operating, maintenance procedures, and forms/checksheets did not have a date of issue or revision. In addition, there was no indication that management had approved a particular procedure or form.

6. 10 CFR 71.135 requires that written records of quality-related activities be maintained for three years after personnel have last engaged in the activity for which the QA program had been issued. It also requires that a record-retention program be established by instruction or procedure.

   Contrary to the above:

   a. The file containing nonconformance and corrective action reports was not available for NRC inspection.

   b. A formal record-filing system was not available. In addition, the responsibility for controlling records was not assigned.

7. 10 CFR 71.137 requires that a comprehensive system of planned and periodic audits be conducted to verify compliance with all aspects of the QA Program. These audits must be performed in accordance with written procedures or checklists by trained personnel.

   JLS&A, "Quality Assurance Program Plan," Revision 3, Annex 1, 1.18 and Annex 2, 2.18, states that pre- and post-audit conferences are to be held, corrective reports written, and timely response and followup action taken to ensure that appropriate corrective action was accomplished.

   Contrary to the above,

   a. The audit file showed no evidence of a planned audit program that identifies the scope of the audit, provisions for pre- and post-audit conferences, with management, regarding the activities under audit, scheduled intervals for auditing of each packaging and transportation activity, or followup of corrective actions taken as a result of the identified program deficiencies.

   b. Documentation of the lead auditor's qualifications was unavailable. The requirements specified in Supplement 2S-3 to ANSI/ASME NQA-1 were not used in qualifying the lead auditors.

LEGEND

FABRICATORS/SUPPLIERS PRESENTATION INSPECTION TREE
PACKAGE SUPPLIER INSPECTION TREE

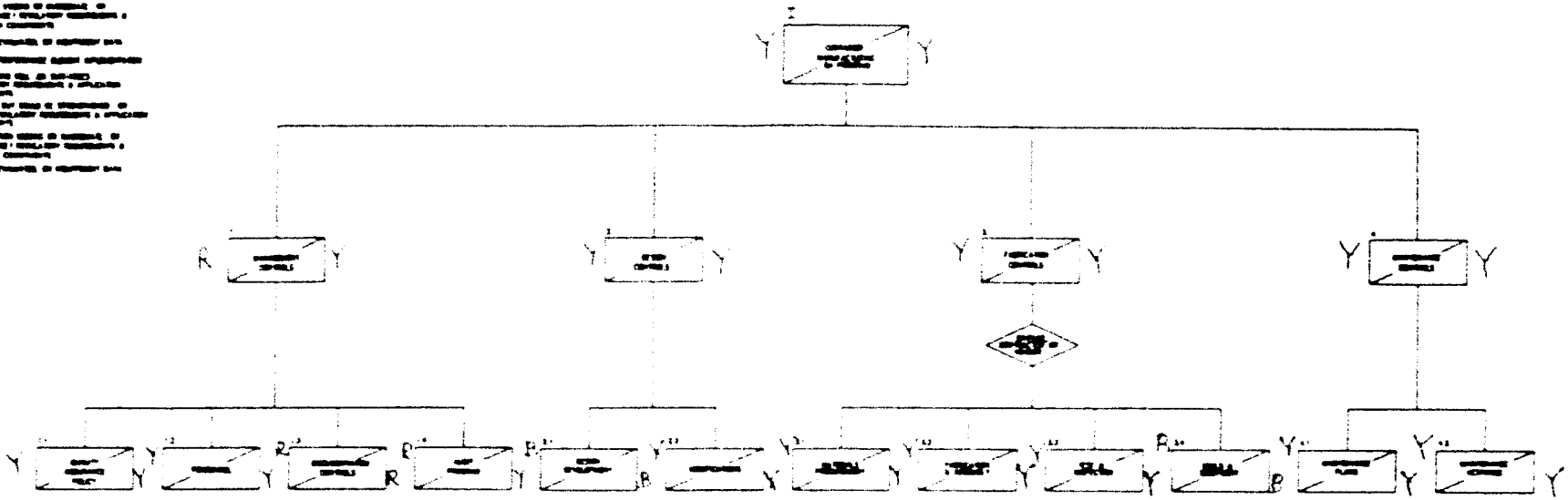


FIGURE 1

J. L. Shepherd & Associates
January 28-31, 1991
Inspection Report No. 710122/90-05

J T Beckham Jr

Georgia Power

HL-1546
001390

March 25, 1991

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

PLANT HATCH - UNITS 1, 2
NRC DOCKETS 50-321, 50-366
OPERATING LICENSES DPR-57, NPF-5
PROPERTY DAMAGE INSURANCE LEVELS

Gentlemen:

In accordance with the requirements of 10 CFR 50.54 (w)(2), Georgia Power Company is submitting the enclosed summary report of the present level and source of on site property damage insurance coverage for the E. I. Hatch Nuclear Plant. The level of coverage maintained is in excess of that required by 10 CFR 50.54 (w)(1).

Should you have any questions in this regard, please contact this office.

Sincerely,

J. T. Beckham, Jr.

JKB/cr

Enclosure: Nuclear Property Insurance Coverages

cc: (See next page.)

1/6

9103290051 910325
PDR ADOCK 05000321
P PDR

A001

Georgia Power

cc: Georgia Power Company
Mr. H. L. Sumner, General Manager - Nuclear Plant
Mr. J. D. Heidt, Manager Engineering and Licensing - Hatch

U.S. Nuclear Regulatory Commission, Washington, D.C.
Mr. K. Jabbour, Licensing Project Manager - Hatch

U.S. Nuclear Regulatory Commission, Region II
Mr. S. D. Ebneter, Regional Administrator
Mr. L. D. Wert, Senior Resident Inspector - Hatch

ENCLOSURE

GEORGIA POWER COMPANY
NUCLEAR PROPERTY INSURANCE COVERAGES
AS OF APRIL 1, 1991
E. I. HATCH NUCLEAR PLANT

PROPERTY - ALL RISK (OPERATING FACILITY)

Policy No.	Insuring Company	Amount of Insurance	Expires
P91-023	Nuclear Mutual Ltd. (NML)	$ 500,000,000	04/01/92
X90-018	Nuclear Electric Insurance Ltd. (NEIL II)	$ 1,125,000,000 Excess $ 500,000,000	11/15/91
X91-156R	American Nuclear Insurers/ Mutual Atomic Energy Liability Underwriters (ANI/MAELU)	$ 700,000,000 Excess $ 500,000,000	01/01/92

DEPARTMENT OF VETERANS AFFAIRS
Royal C Johnson Veterans Memorial Hospital and Regional Office Center
PO Box 5046, 2501 West 22nd Street
Sioux Falls SD 57117

February 20, 1991

438/115

Director, Nuclear Medicine Service (141A5)
Department of Veterans Affairs
810 Vermont Ave., N.W.
Washington, D.C. 20420

SUBJ: Response to NRC Notice of Violation
NRC license #40-16336-01

Following is our reply to each violation stated in the Notice of Violation dated January 30, 1991, resulting from an NRC Inspection November 26-27, 1990:

A. Violation #1: Dose calibrator accuracy records did not reflect model and serial # of the dose calibrator, check source and RSO signature.

Violation #2: Linearity tests did not reflect model and serial number of the dose calibrator and the RSO signature.

Violation #3: Leak tests did not reflect model and serial numbers of each source tested and the RSO signature.

B. Response #1: Admittedly the accuracy, linearity and leak tests did not reflect the model and serial numbers and did not include the RSO signature as indicated in the Notice of Violation.

Response #2: Corrective steps taken include the addition of model and serial numbers of the dose calibrator and sources used on accuracy, linearity and leak test records. The RSO has reviewed and signed accuracy, linearity and leak test records. Results achieved include test records in compliance with our NRC license conditions.

Response #3: Preventive measures taken to avoid further violations include review of our license conditions with attention to dose calibrator calibrations in Reg. guide 10.8 Rev. 2, Appendix C, and implementation of the procedures.

Response #4: Full compliance will be achieved by March 1, 1991, when the accuracy, linearity and leak tests will be completed, reviewed and signed by the RSO.

Response #5: Dose calibrator violations identified in the 1988 inspection were corrected immediately following the inspection. Dose calibrator violations identified in the 1990 inspection included model and serial number identification of sources and equipment used. These will have been corrected by the first of March, 1991.

91-601

Document contains
illegible data

As stated above preventive measures against further dose calibration violations include a thorough review and implementation of dose calibration procedures as stated in Reg. Guide 10.8, Rev. 2, Appendix C.

C. Comments: We would like to thank Selvan Rajendram, the inspector, for the courteous helpful and professional manner in which he conducted the inspection and exit interview. Mr. Rajendram not only addressed the three minor recommendations but also spoke highly of our byproduct materials usage program. We appreciated hearing the positive aspects of our program.

R. VINCENT CRAWFORD
Center Director

cc:
Regional Director, Central Region (III)
Regional Director, Ann Arbor (I.C.)

VAMC, LEAVENWORTH KS 66048

February 25, 1991 686/115

Regional Director, Central Region (132/141A5)
Department of Veterans Affairs
810 Vermont Avenue, NW
Washington, DC 20420

SUBJ: Amend Radioactive Materials License #15-08114-01

1. We need to amend the Radioactive Materials License #15-08114-01 issued to the DDE VAMC, Leavenworth, Kansas. We need Jafar Amini, M.D., listed as an authorized user on our Radioactive Materials License. Dr. Amini should be licensed for imaging and localization studies in accordance with 35.920 (A-2). A copy of Dr. Amini's American Board of Radiology certificate is enclosed. Also, please delete Dr. Jerry Everett as an authorized user on out license.

2. If you have any questions concerning this request, please contact Dr. M.G. Rao at FTS 752-0482. Thank you for your cooperation.

JAMES H. CUER
Medical Center Director

Enclosure

cc: RD(132), Ann Arbor, MI

The American Board of Radiology

Organized through the cooperation of the
American College of Radiology, the American Roentgen Ray Society,
the American Radium Society, the Radiological Society of North America,
the Section on Radiology of the American Medical Association
and the American Society of Therapeutic Radiologists
Hereby certifies that

Jafar Amini, M.D.

Has pursued an accepted course of graduate study
and clinical work, has met certain standards and qualifications and
has passed the examinations conducted under the authority of
The American Board of Radiology
On this tenth day of December, 1976
Thereby demonstrating to the satisfaction of the Board
that he is qualified to practice the specialty of

Radiology

THE AMERICAN BOARD OF RADIOLOGY
INCORPORATED IN THE DISTRICT OF COLUMBIA
1934

Sidney W. Nelson
President

C. Allen Good
Secretary

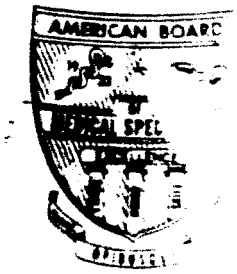


EXHIBIT 2

SUPPLEMENT A

SUPPLEMENT — U.S. NUCLEAR REGULATORY COMMISSION

TRAINING AND EXPERIENCE
AUTHORIZED USER OR RADIATION SAFETY OFFICER

1. NAME OF PROPOSED AUTHORIZED USER OR RADIATION SAFETY OFFICER	2. FOR PHYSICIANS, STATE OR TERRITORY WHERE LICENSED
JAFAR AMINI, M.D.	

3. CERTIFICATION

SPECIALTY BOARD A	CATEGORY B	MONTH AND YEAR CERTIFIED C
American Board of Radiology (copy of license enclosed)		

4. TRAINING RECEIVED IN BASIC RADIOISOTOPE HANDLING TECHNIQUES

FIELD OF TRAINING A	LOCATION AND DATE(S) OF TRAINING B	TYPE AND LENGTH OF TRAINING	
		CLOCK HOURS IN LECTURE OR LABORATORY	CLOCK HOURS OF SUPERVISED ON-THE-JOB EXPERIENCE
a. RADIATION PHYSICS AND INSTRUMENTATION			
b. RADIATION PROTECTION			
c. MATHEMATICS PERTAINING TO THE USE AND MEASUREMENT OF RADIOACTIVITY			
d. RADIATION BIOLOGY			
e. RADIOPHARMACEUTICAL CHEMISTRY			

5. EXPERIENCE WITH RADIATION. *(Actual use of Radioisotopes or Equivalent Experience)*

ISOTOPE	mCi USED AT ONE TIME	LOCATION	CLOCK HOURS	TYPE OF USE
			MAF	

Indian Point 3
Nuclear Power Plant
P.O Box 215
Buchanan, New York 10511

914-736-8000

New York Power Authority

March 12, 1991
IP3-91-021
IPO-91-043L

Docket No. 50-286
License No. DPR-64

Mr. Thomas T. Martin
Region I Administrator
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Dear Mr. Martin:

In accordance with 10CFR50.4(b)(5), one (1) copy of the most recent revisions to our Emergency Plan/Procedures has been sent to the NRC Document Control Desk in Washington, D.C. Two (2) copies have been sent to your office, Emergency Preparedness Section, and one (1) copy has been sent to the IP-3 Resident Inspector.

These changes have been evaluated against 10CFR50.54(q) and have been determined not to decrease the effectiveness of the Indian Point 3 Emergency Plan. The Plan and Procedures, as changed, continue to meet the standards of 10CFR50.47 and Appendix E.

Sincerely,

Joseph E. Russell
Resident Manager
Indian Point No. 3 Nuclear Power Plant

JER/DB/lat

cc: Document Control Desk (original)
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Resident Inspector's Office
Indian Point No. 3
U.S. Nuclear Regulatory Commission
P.O. Box 337
Buchanan, New York 10511

9103290018 910312
PDR ADOCK 05000286
F PDR

MAR 0 6 1991

Docket No. 030-17770 License No. 20-13627-02

General Foods Corporation
Atlantic Gelatin Corporation
ATTN: Joseph Saladini
Hill Street
Woburn, Massachusetts 01801

Gentlemen:

Subject: Inspection No. 030-17770/91-001

This refers to your letter dated January 29, 1991, in response to our letter dated January 24, 1991.

Thank you for providing the information documented in your letter. If the license reviewer has any questions regarding the termination request, then he will contact you.

Your cooperation with us is appreciated.

Sincerely,

Original Signed By:
John D. Kinneman

John D. Kinneman, Chief
Nuclear Materials Safety Section B
Division of Radiation Safety
and Safeguards

cc:
Public Document Room (PDR)
Nuclear Safety Information Center (NSIC)
Commonwealth of Massachusetts

bcc:
Region I Docket Room (w/concurrences)

RI:DRSS	RI:DRSS
Courtemanche/bj	Kinneman
02/ /91	03/ /91

OFFICIAL RECORD COPY RL GENERAL FOODS - 0001.0.0
02/28/91

Return Original to Region I IE:07

9103220014 910306
REG1 LIC30
20-13627-02 PDR

KRAFT GENERAL FOODS

Joseph J. Saladin

9103290015 910306
REG1 LIC30
20-13627-02 PDR

NRC FORM 314
(5 90)
10 CFR 30 36(c)(1)(iv)
10 CFR 40 42(c)(1)(iv)
10 CFR 70 38(c)(1)(iv)

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF DISPOSITION OF MATERIALS

INSTRUCTIONS SEND THE COMPLETED CERTIFICATE TO THE NRC OFFICE SPECIFIED ON THE REVERSE. *(All items MUST be completed--print or type)*

APPROVED BY OMB 3150-0028
EXPIRES 4/30/92

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST 30 MIN FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714) U S NUCLEAR REGULATORY COMMISSION WASHINGTON DC 20555 AND TO THE PAPERWORK REDUCTION PROJECT (3150-0028) OFFICE OF MANAGEMENT AND BUDGET WASHINGTON DC 20503

LICENSEE NAME AND ADDRESS

GENERAL FOODS CORPORATION
ATLANTIC GELATIN DIVISION
HILL STREET
WOBURN, MASSACHUSETTS 01801

LICENSE NUMBER

[illegible]

LICENSE EXPIRATION DATE

JAN 31, 19[illegible]

THE LICENSEE OR ANY INDIVIDUAL EXECUTING THIS CERTIFICATE ON BEHALF OF THE LICENSEE CERTIFIES THAT
(Check and/or complete the appropriate item(s) below.)

A. MATERIALS DATA *(Check one and complete as necessary.)*

☐ 1 NO MATERIALS HAVE EVER BEEN PROCURED OR POSSESSED BY THE LICENSEE UNDER THIS LICENSE

OR

☒ 2 ALL MATERIALS PROCURED AND/OR POSSESSED BY THE LICENSEE UNDER THE LICENSE NUMBER CITED ABOVE HAVE BEEN DISPOSED OF IN THE FOLLOWING MANNER *(If additional space is needed, use the reverse side or provide attachments.)*

Describe specific material transfer actions and, if there were radioactive wastes generated in terminating this license, the disposal actions, including the disposition of low level radioactive waste, mixed waste, Greater than Class C waste, and sealed sources, if applicable

One electron capture detector from L. Perkin Elmer GC
Model 8500[illegible], containing one 15mCi Ni-63 sealed source,
serial no. 1136

For transfers, specify the date of the transfer, the name of the licensed recipient, and the recipient's NRC license number or Agreement State name and license number

NSSI/Sources and Services, Incorporated, via [illegible]
Texas Dept. of Health License No. [illegible]

If materials were disposed of directly by the licensee rather than transferred to another licensee, licensed disposal site or waste contractor, describe the specific disposal procedures *(e.g., decay in storage)*

B. OTHER DATA

☒ 1 OUR LICENSE HAS NOT YET EXPIRED. PLEASE TERMINATE IT

2 WAS A RADIATION SURVEY CONDUCTED TO CONFIRM THE ABSENCE OF LICENSED RADIOACTIVE MATERIALS AND TO DETERMINE WHETHER ANY CONTAMINATION REMAINS ON THE PREMISES COVERED BY THE LICENSE? *(Check one)*

☒ NO *(Attach explanation)*

☐ YES, THE RESULTS *(Check one)*

- ☐ ARE ATTACHED, or
- ☐ WERE FORWARDED TO NRC ON *(Date)*

3. THE PERSON TO BE CONTACTED REGARDING THE INFORMATION PROVIDED ON THIS FORM

NAME: Joseph J. Saladini | TELEPHONE NUMBER: [illegible]

4 MAIL ALL FUTURE CORRESPONDENCE REGARDING THIS LICENSE TO

Atlantic Gelatin Corporation Attn: J. Saladini
Hill Street
Woburn, Massachusetts 01801

CERTIFYING OFFICIAL

I CERTIFY UNDER PENALTY OF PERJURY THAT THE FOREGOING IS TRUE AND CORRECT

SIGNATURE: Joseph J. Saladini | DATE: 1-18-91

PRINTED NAME AND TITLE: Joseph J. Saladini, Group Leader Tech. Services Dept.

WARNING: FALSE STATEMENTS IN THIS CERTIFICATE MAY BE SUBJECT TO CIVIL AND/OR CRIMINAL PENALTIES. NRC REGULATIONS REQUIRE THAT SUBMISSIONS TO THE NRC BE COMPLETE AND ACCURATE IN ALL MATERIAL RESPECTS. 18 U.S.C. SECTION 1001 MAKES IT A CRIMINAL OFFENSE TO MAKE A WILLFULLY FALSE STATEMENT OR REPRESENTATION TO ANY DEPARTMENT OR AGENCY OF THE UNITED STATES AS TO ANY MATTER WITHIN ITS JURISDICTION.

NRC FORM 314 (5 90)

9102290016 910205
REG1 LIC30
20-13627-02 FDR

MATERIALS LICENSE

Amendment No. 01

Pursuant to the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974 (Public Law 93-438), and Title 10, Code of Federal Regulations, Chapter I, Parts 30, 31, 32, 33, 34, 35, 40 and 70, and in reliance on statements and representations heretofore made by the licensee, a license is hereby issued authorizing the licensee to receive, acquire, possess, and transfer byproduct, source, and special nuclear material designated below; to use such material for the purpose(s) and at the place(s) designated below; to deliver or transfer such material to persons authorized to receive it in accordance with the regulations of the applicable Part(s). This license shall be deemed to contain the conditions specified in Section 183 of the Atomic Energy Act of 1954, as amended, and is subject to all applicable rules, regulations and orders of the Nuclear Regulatory Commission now or hereafter in effect and to any conditions specified below.

Licensee	
1. General Foods Corporation Atlantic Gelatin Division	In accordance with letter dated June 25, 1985, 3. License number 20-13627-02 is amended in its entirety to read as follows:
2. Hill Street Wouburn, Massachusetts 01801	4. Expiration date January 31, 1991
	5. Docket or Reference No. 030-17770

6. Byproduct, source, and or special nuclear material	7. Chemical and/or physical form	8. Maximum amount that licensee may possess at any one time under this license
A. Nickel 63	A. Foils in Perkin-Elmer Model 330-0119 detector cells	A. Not to exceed 15 millicuries per foil

9. Authorized use

A. For use in Perkin-Elmer Sigma Series gas chromatographs for sample analysis

CONDITIONS

10. Licensed material shall be used only at the licensee's facilities at Hill Street Woburn, Massachusetts.

11. The licensee shall comply with the provisions of Title 10, Chapter 1, Code of Federal Regulations, Part 19, "Notices, Instructions, and Reports to Workers; Inspections" and Part 20, "Standards for Protection Against Radiation."

12. Licensed material shall be used by, or under the supervision of, Norman E. Mahoney or Joseph J. Saladini.

13. In lieu of using the conventional radiation caution colors (magenta or purple on yellow background) as provided in Section 20.203(a)(1), Title 10, Code of Federal Regulations, Part 20, the licensee is hereby authorized to label detector cells and cell baths, containing licensed material and used in gas chromatography devices, with conspicuously etched or stamped radiation caution symbols without a color requirement.

14. A. Each chromatograph detector containing Nickel 63 shall be tested for leakage and/or contamination at intervals not to exceed six months. In the absence of a certificate from a transferor indicating that a test has been made within six months prior to the transfer, a detector received from another person shall not be put into use until tested.

MATERIALS LICENSE
SUPPLEMENTARY SHEET

License number
20-13627-02
Docket or Reference number
030-17770
Amendment No. 01

(14. continued) CONDITIONS

B. The test shall be capable of detecting the presence of 0.005 microcurie of radioactive material on the test sample. The test sample shall be taken from the surfaces of the device in which the foil is mounted or stored on which one might expect contamination to accumulate. Records of leak test results shall be kept in units of microcuries and maintained for inspection by the Commission.

C. If the test reveals the presence of 0.005 microcurie or more of removable contamination, the licensee shall immediately withdraw the foil from use and shall cause it to be decontaminated and repaired or to be disposed of in accordance with Commission regulations. A report shall be filed within 5 days of the test with the U.S. Nuclear Regulatory Commission, Region I, 631 Park Avenue, King of Prussia, Pennsylvania 19406, describing the equipment involved, the test results, and the corrective action taken.

D. The licensee is authorized to collect leak test samples in accordance with the procedures described in the licensee's application dated July 18, 1980 for analysis by the manufacturer. Alternatively, leak test samples may be collected and/or analyzed by other persons specifically authorized by the Commission or an Agreement State to perform such services.

15. The licensee shall conduct a physical inventory every six (6) months to account for all foils received and possessed under the license. The records of the inventories shall be maintained for two (2) years from the date of the inventory for inspection by the Commission, and shall include the quantities and kinds of licensed material, location of the foils and the date of the inventory.

16. Detector cells containing licensed material shall not be opened or the foil sources removed from the detector cell by the licensee.

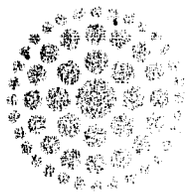
17. Except as specifically provided otherwise by this license, the licensee shall possess and use licensed material described in Items 6, 7, and 8 of this license in accordance with statements, representations, and procedures contained in letter dated June 25, 1985, and application dated July 18, 1980. The Nuclear Regulatory Commission's regulations shall govern the licensee's statements in applications or letters, unless the statements are more restrictive than the regulations.

For the U.S. Nuclear Regulatory Commission

Date JAN 31 1996

By Original Signed By:
John D. Kinneman

Nuclear Materials Safety and
Safeguards Branch, Region I
King of Prussia, Pennsylvania 19406



Florida Power
CORPORATION
Crystal River Unit 3
Docket No. 50-302

March 22, 1991
3F0391-15

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Subject: Updated Design Information for the Auxiliary Feedwater Pump Addition

Reference: A. FPC letter dated May 31, 1990
B. NRC letter dated September 19, 1990

Dear Sir:

Florida Power Corporation (FPC) is submitting this letter to provide the NRC with updated design information about the auxiliary feedwater (AFW) pump addition at Crystal River Unit 3 (CR-3). We are providing revisions to certain parts of Reference A. These changes are not expected to alter the conclusions in your letter of September 19, 1990.

Since the conceptual design was submitted in the reference letter, FPC has evaluated the safety significance and cost benefit of the new AFW pump being totally initiated and controlled from the main control room. FPC has decided to proceed with a design in which the AFW pump is normally isolated from the Emergency Feedwater (EF) System with closed manually operated isolation valves. The installation of this AFW pump addition remains planned for Refuel 8, now scheduled for the Spring of 1992.

Reference A contained 5 attachments describing the project, conceptual drawings including preliminary piping layouts, and a schedule of activities. Attachments 1 and 2 to this letter replace Attachments 1 and 2 in the reference letter.

1/1

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PDR ADOCK 05000302
P PDR

A001

Changes to the Project Description (Attachment 1) are indicated by a black line in the right margin. Attachments 3 and 4 of the reference letter are not being updated with this submittal because they are conceptual piping layout drawings and this design revision is adequately shown in the schematic (Attachment 2). Attachment 5 is a bar chart with schedular information showing completion of the installation in Refuel 8. This schedule has not changed.

Sincerely,

P. M. Beard, Jr.
Senior Vice President
Nuclear Operations

PMB/JWT

Attachments

xc: Regional Administrator, Region II
Senior Resident Inspector
NRR Project Manager

AUXILIARY FEEDWATER PUMP ADDITION TO THE FEEDWATER SYSTEM

PROJECT DESCRIPTION

BACKGROUND

This project will install an additional non-safety-related source of feedwater to reduce reliance on long-term High Pressure Injection/Power Operated Relief Valve (HPI/PORV) cooling should a loss of all main and emergency feedwater occur. FPC's philosophy for this additional source of feedwater is to install a system which is reliable, simple to operate, and maintains the integrity of the Emergency Feedwater (EF) System which it is intended to supplement. To assure that reliability is maintained, the AFW design will provide accessibility to several on-site sources of feedwater and the AFW pump will be specified using a recognized industry standard which produces reliable commercial grade feedwater pumps. All AFW operation will be accomplished from the main control room, except for manually opening/closing two isolation valves in the Intermediate Building. The system will be simple to operate by remote-manual controls located on the same main control room panel as the EF System and independent of the Emergency Feedwater Initiation and Control (EFIC) System. The integrity of the EF System will be maintained by connecting the AFW supply to the emergency feedwater lines downstream of the EF System vector control valves, and isolating this interconnection with closed manual valves. The AFW addition will also utilize the same upper feedwater nozzles as the EF System.

DESIGN BASES

1. The design of the AFW pump, designated FWP-7, and associated equipment is based on the following sequence of events:

   a. A loss of main feedwater (MFW) occurs as the initiating event.

   b. Both trains of the Emergency Feedwater System fail to supply water to the steam generators.

   c. The AFW pump addition is initiated.

   d. Once the AFW pump addition is started, the steam generator level is manually controlled by the control room operator.

2. The AFW addition is designed to provide an additional non-safety grade method of secondary heat removal, in order to reduce reliance on long term HPI/PORV cooling following a loss of all main and emergency feedwater. The AFW addition has been sized to reduce the time that HPI/PORV cooling is relied upon after a loss of all feedwater. When the AFW addition is started, it can provide up to 250 gpm of auxiliary feedwater to the upper nozzles of each steam generator at maximum system pressure.

3. Following the failure of both trains of the Emergency Feedwater System, no subsequent failures are assumed to occur on any safety-related component for the remainder of the event, including an automatic/spurious restart of the Emergency Feedwater System after the AFW addition has been initiated.

4. The AFW addition is designed to withstand the spurious or inadvertent actuation of the AFW Pump and control valves. The pump will normally be off and the control valves will normally be closed. Spurious actuation of one or all of these components will not admit water to the steam generators since normally closed manual valves FWV-222 and FWV-223 will prevent this flow. In addition, AFW control valves FWV-216 and FWV-217 are designed to fail closed on a loss of instrument air or electric power.

5. Piping and valves within the Intermediate Building will be Seismic Category I. AFW piping, valves, and components located outside the Intermediate Building will be Seismic Category III, non-safety-related. Piping will be designed to ANSI B31.1-1967.

6. The AFW addition will have high reliability. The design minimizes sharing of equipment with the existing Emergency Feedwater System, and also minimizes the number of active components required to operate to supply auxiliary feedwater to the steam generators.

7. FWP-7 and AFW control valves FWV-216/217 will be designed for remote manual actuation from the Main Control Room. The AFW addition will be completely independent of EFIC, which controls the Emergency Feedwater System.

8. No credit will be taken for this subsystem to meet fire protection requirements. The AFW design will ensure that existing fire protection features will not be degraded.

9. Since the AFW addition will operate only under emergency conditions, High Energy Line Break (HELB) rupture protection is not a required. Existing HELB protection will not be degraded.

10. FPC is providing protection (e.g. raised pedestals) for AFW equipment against the flooding resulting from a main feedwater line break, since such an event could create a need for AFW. Complete protection against other internal flooding events is not being provided.

11. Electrical and control systems will not be Class 1E.

DESIGN FEATURES

1. FWP-7 and most of the associated piping will be located in the Turbine Building, with the remainder of the piping located in the Intermediate Building. The layout of the equipment and piping is shown on the Attachment 2, CRYSTAL RIVER 3 FWP-7 CONCEPTUAL DESIGN Flow Schematic.

2. FWP-7 will normally be aligned to take suction from the existing Condensate Storage Tank, CDT-1, as the primary water source. The Condenser Hotwell and the Emergency Feedwater Tank, EFT-2, will be available as backup water sources.

3. The design will include a minimum flow recirculation line which will be sized for approximately 240 gpm and a full flow recirculation test line. The minimum flow recirculation line will not be isolated while feeding the generators.

4. FWP-7 will be capable of delivering a minimum of 500 gpm to the steam generators (250 gpm to each steam generator) at maximum system pressure. Feedwater will be injected into each steam generator using the upper nozzles normally used by the Emergency Feedwater System.

5. In general, FWP-7 is being specified to the requirements of API-610, February 1989, "Centrifugal Pumps for General Refinery Service". To the extent practical, the AFW pump and motor will be self-contained, with little or no reliance on external systems for cooling, lubrication, etc.

6. The motor for FWP-7 will be fed from new non-Class 1E 4160V switchgear. This switchgear will be powered from the existing 6900V Reactor Auxiliary Bus 3A which is not capable of being connected to the emergency diesel generators.

7. Flow will be independently regulated to each steam generator by new air-operated control valves, designated FWV-216 and FWV-217.

8. FWP-7 and the flow control valves will be remote manually operated from the main control room. The pump and valve controls will be located on the main control board near the controls for the Emergency Feedwater System and will consist of the following:

   a. FWP-7 control switch.

   b. FWV-216/FWV-217 control switches (2 total).

   c. Flow indication to each steam generator.

9. A local control station will be provided for FWP-7 and local control of FWV-216 and FWV-217 will be by manual handwheel only. Local flow indication only will be provided for the minimum and full flow recirculation lines.

OPERATION

1. The AFW pump will be used after a failure of both Emergency Feedwater System trains. AFW will be selected after the Emergency Feedwater System since it must be manually initiated.

2. The AFW addition is not designed for use in plant start-up or normal plant operations.

3. To supply auxiliary feedwater to the steam generators will require only three actions:

   a. Start the AFW pump (on recirculation) from the main control room.

   b. Open manual isolation valves FWV-222 and FWV-223 located in the Intermediate Building.

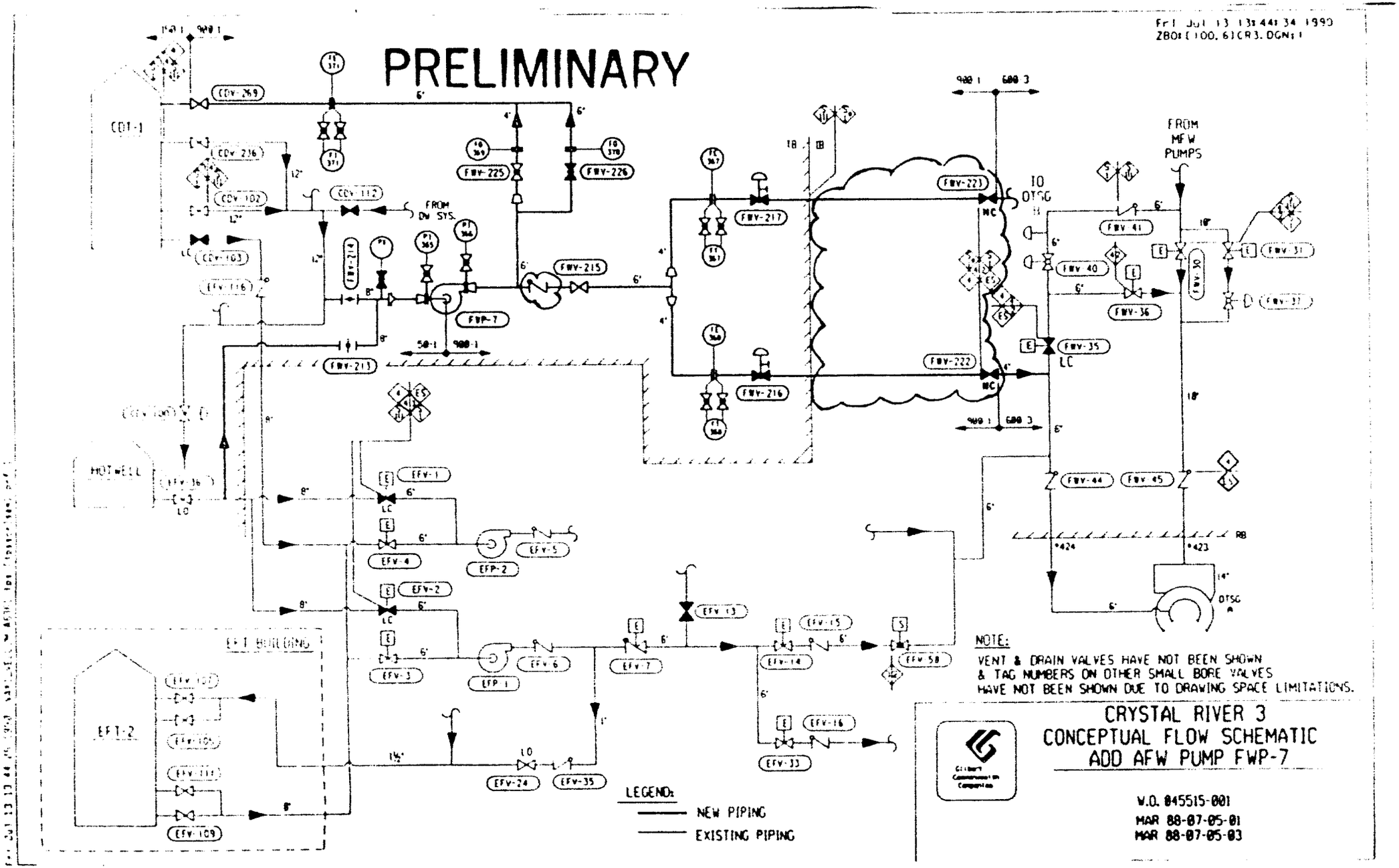
   c. Open/regulate the AFW flow control valves from the main control room, based on observed flow indication and steam generator level.

4. Provisions will be made for periodic testing of the AFW pump.

5. The AFW addition and the Emergency Feedwater System will not be intentionally operated together to feed the steam generators, due to upper nozzle flow limits. If simultaneous AFW addition/Emergency Feedwater System operation occurs as a result of spurious or inadvertent actuation, the control room operator will be procedurally instructed to secure one of these sources.

ATTACHMENT 2 TO 3F0391-15

INDICATES CHANGES FOR RE-DESIGN TO MANUALLY ISOLATED AFW SOURCE



SGTB:GMB
71-0416

Thiokol Wasatch Division
ATTN: Arnold Peart, RSO
P.O. Box 707, M/S A-91
Brigham City, UT 84302-0707

Gentlemen:

As requested in your telephone conversation of March 21, 1991 with John Cook of my office, Quality Assurance Approval No. 0416 has been terminated.

Please advise us of the location at which the records required by 10 CFR Part 71.91 are to be retained.

Sincerely,

/S/ John P. Cook for
Charles E. MacDonald, Chief
Transportation Branch
Division of Safeguards and
Transportation, NMSS

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NAME:	GMBennington	LLGordon	~~JRCook~~	CEMacDonald
DATE:	03/27/91	03/27/91	03/27/91	03/27/91

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March 20, 1991

PG&E Letter No. DCL-91-057

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Re: Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
Diablo Canyon Units 1 and 2
10 CFR 50.59 Annual Report of Facility Changes, Procedure Changes, Tests, and Experiments During March 23, 1989 - March 22, 1990

Gentlemen:

Pursuant to 10 CFR 50.59, enclosed is the Annual Report of Changes, Tests, and Experiments for Diablo Canyon Power Plant (DCPP) Units 1 and 2 for the reporting interval of March 23, 1989 to March 22, 1990. This is the same as the reporting interval of the annual Final Safety Analysis Report (FSAR) Update revision.

Licensees are allowed by 10 CFR 50.59 to make changes in the facility as described in the FSAR, make changes in procedures as described in the FSAR, and conduct tests and experiments not described in the FSAR without prior NRC approval, provided that the changes, tests, or experiments do not involve changes in the Technical Specifications or unreviewed safety questions. Such changes are required to be reported on an annual basis by 10 CFR 50.59(b)(2).

Proposed facility changes, procedure changes, tests, or experiments that are determined to involve unreviewed safety questions or changes to the Technical Specifications are submitted separately to the NRC for prior approval under the provisions of 10 CFR 50.90.

Changes in the Facility as Described in the FSAR. The enclosed Annual Report provides a brief description of the 10 CFR 50.59 facility design changes, including a summary of each 10 CFR 50.59 evaluation. Each change is complete and has been accepted by the plant staff. Each evaluation was reviewed by the Plant Staff Review Committee (PSRC) and approved by the Plant Manager.

None of the design changes involved an unreviewed safety question or a change to the DCPP Technical Specifications, as determined by the PSRC. The changes do not (a) increase the probability or consequences of an accident or malfunction important to safety as previously analyzed in the FSAR; (b) create the possibility of an accident or malfunction of equipment not previously analyzed in the FSAR; or (c) reduce the margin of safety as defined in the bases of the Technical Specifications.

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Design changes resulting in minor changes to FSAR Update figures (e.g., piping and instrumentation drawings and schematics) that do not impact the safety of operations are not included in the Annual Report.

Changes in Procedures as Described in the FSAR Update. The enclosed Annual Report provides a brief description of a 10 CFR 50.59 procedure change, including a 10 CFR 50.59 evaluation summary. The procedure change does not involve an unreviewed safety question or a change to the DCPP Technical Specifications, as determined by the PSRC.

Tests or Experiments Not Described in the FSAR Update. Tests and experiments are performed in accordance with procedures. The enclosed Annual Report also provides a brief description of tests or experiments not described in the FSAR Update together with a 10 CFR 50.59 evaluation summary. None of the tests or experiments involved an unreviewed safety question or a change to the DCPP Technical Specifications, as determined by the PSRC.

Jumpers/Lifted Leads. Several leads that were lifted during the time period covered by the Annual Report were determined to result in a change to a function of a system operation as described in the FSAR Update. A brief discussion and 10 CFR 50.59 evaluation summary for each of the lifted leads is included in the Annual Report. None of the lifted leads involved an unreviewed safety question or a change to the DCPP Technical Specifications, as determined by the PSRC.

Sincerely,

J. D. Shiffer

cc: Ann P. Hodgdon
John B. Martin
Phillip J. Morrill
Paul P. Narbut
Harry Rood
CPUC
Diablo Distribution

Enclosure

5244S/0085K/RWM/1392

ENCLOSURE

10 CFR 50.59 ANNUAL REPORT OF FACILITY CHANGES, PROCEDURE CHANGES, TESTS, AND EXPERIMENTS

MARCH 23, 1989 - MARCH 22, 1990

DIABLO CANYON POWER PLANT
UNITS 1 AND 2

DOCKET NOS. 50-275 AND 50-323

Pacific Gas and Electric Company

A. Facility Changes

No.	Description	Unit Applicability	Identification
1.	Add Bearing Lift System for Low Pressure Turbine "C" Rotor Bearing No. 8	2	DCP M-16286 R.2
2.	Remove Alarms from Reactor Protection System Cabinets' (Hagen Racks') Doors	1,2	DCP E-17759 R.1 E-18759 R.1
3.	Provide Concrete Pad for the Mobile Radwaste Processing System	1,2	DCP M-29953 R.10
4.	Abandon/Lay-up Radioactive Waste Concentrator Package	1,2	DCP M-31188 R.9
5.	Revise Fire Detection and Suppression Controls for the Emergency Diesel Generator Rooms	1,2	DCP M-33398 R.1 M-34398 R.1
6.	Permit Replacement of Barton with Rosemount Pressure, Level and dP Transmitters	1,2	DCP J-33589 R.0 J-34589 R.0
7.	Install two, new, Liquid Radioactive Waste After-Filters	1,2	DCP M-35397 R.1
8.	Convert the Waste Concentrates Holding Tank to a Second Boric Acid Concentrates Holding Tank	1,2	DCP M-39457 R.0
9.	Modify the Reactor Cavity Manipulator Crane Controls	1	DCP M-39515 R.0
10.	Cut and Cap Spare Reactor Vessel CRDM Head Adaptors	1	DCP M-39666 R.1
11.	Install Digital Feedwater Control System	1,2	DCP J-41250 R.1 J-41373 R.2 E-41636 R.0 J-42250 R.1 J-42373 R.1 E-42636 R.0
12.	Lock Closed the Primary Water Makeup Supply to the CCW Surge Tank	1,2	DCP M-41291 R.0 M-42291 R.0
13.	Disconnect Radiation Monitor RE-20 from the Radiation Monitoring System	1,2	DCP J-41463 R.1
14.	Remove Moderate Energy Line Break Spray Guards providing protection for the Residual Heat Removal Pump Motors	1	DCP M-41490 R.0

A. Facility Changes (cont'd)

No.	Description	Unit Applicability	Identification
15.	Disconnect the Plant Process Computer from Vital Power Supplies	1,2	DCP J-41527 R.0 J-42527 R.0
16.	Install Air Conditioning System for the Plant Process Computer Room	1	DCP J-41528 R.1
17.	Disconnect and Remove Power Range Neutron Flux Recorders	1,2	DCP J-41569 R.1 J-42569 R.2
18.	Install Quick Opening Fuel Transfer Tube Closure Assembly	1,2	DCP M-41650 R.0 M-42650 R.0
19.	Replace the Sequence of Events Recorder Subsystem of the Main Annunciator System	1,2	DCP E-41723 R.0 E-42352 R.0 J-42534 R.0
20.	Reconfigure/Upgrade Seismic Trip System	1,2	DCP J-41746 R.1 J-43064 R.4 J-42746 R.1 J-44064 R.0
21.	Install Cathode Ray Tubes and Personal Computers in the Control Console in the Main Control Room	2	DCP J-43013 R.0 J-42534 R.0
22.	Raise the Condensate Storage Tank Low-Low Level Alarm Setpoint	1,2	DCP J-43013 R.0 J-44013 R.0
23.	Install Fiberoptic Inspection Ports in the Main Steam Isolation Valves	1,2	DCP P-43028 R.0 P-44028 R.0
24.	Install Nitrogen Supply System for the I&C Shop Calibration Lab	1,2	DCP A-43141 R.0
25.	Install new Reactor Vessel Stud Tensioner Hoists	1,2	DCP M-43298 R.0 M-44298 R.0
26.	Alter and Document the Configuration of the Containment Recirculation Sump	1,2	DCP C-43642 R.0 C-43762 R.1 C-44642 R.0 C-46080 R.0
*27.	Disconnect Air Supply through Containment Penetration 80 and Cap the Penetration	1,2	DCN 0-EM-1106 R.0
*28.	Remove Radwaste Solidification/ Drumming/Storage Facility from Fuel Handling Building	1,2	DCN 2-EM-1257 R.0

A. Facility Changes (cont'd)

No.	Description	Unit Applicability	Identification
*29.	Replace Containment Electrical Penetration Assemblies 21E and 39E	1,2	DCN 1-EC-19688 R.4 1-EE-19688 R.1 2-EC-20688 R.1 2-EE-20688 R.0

* These changes were completed in previous reporting periods but were not included in earlier Annual 50.59 Reports; they are included in this report to update the reporting record.

B. Procedure Changes

No.	Description	Unit Applicability	Identification
1.	A Temporary Change to reduce Tavg 5 degrees for End of Life, Cycle 3 operation	1	OP L-4

C. Procedures (Test)

No.	Description	Unit Applicability	Identification
1.	Procedure for operational testing of the Auxiliary Building Supply and Exhaust Fan Inlet Damper Vanes to obtain fan flow switch d/p data and to verify that replacement controllers for the inlet vanes perform per design	2	SU 23.3A7, Rev. 0
2.	Procedure for functionally testing CVI System after implementation of an approved facility change which provides an alternate actuation circuit and 120 Vac power source	1	SU 35.6 A2, Rev. 0
3.	One time test of the Steam Dump Turbine Trip Controller actuation which was changed from actuation by SSPS slave relays to actuation by Reactor Trip relays RTK A/B. The new actuation is the result of the implementation of an approved design change	1	TB-8934, Rev. 0
4.	Temporary (test) Procedure to verify the post-replacement wiring and correct operation SSPS slave relay test switches	1	TB-8929, Rev. 0

C. Procedures (Test) (cont'd)

No.	Description	Unit Applicability	Identification
5.	Procedure to determine the margin of voltage and current in the MSIV solenoid continuity test circuit. Pickup voltage and current is measured for each MSIV solenoid to determine if the dropping resistors in the continuity test circuit are properly sized	1,2	TB-8905, Rev. 0
6.	Procedure to determine the margin of margin of voltage and current in the Anti-Mortoring and Sequencial Trip Relay test circuits. The data obtained will be used to determine if the dropping resistors in the continuity test circuits are properly sized	1,2	TB-8906, Rev. 0
7.	Procedure to measure the pickup voltage and current for the Turbine Trip Solenoids. The data obtained will be used to determine if the dropping resistors in continuity test circuits are properly sized	1,2	TB-8909, Rev. 0
8.	Procedure provides a leak test of Spray Additive Tank out line check valves 8998A and 8998B	1,2	STP V-21, Rev. 0
9.	Procedure pressurizes the Boron Injection Tank discharge piping to a nominal RCS pressure in order that an an external inspection of the piping may be accomplished	1	STP M-112, Rev. 0

D. Jumpers/Lifted Leads

No.	Description	Unit Applicability	Identification
1.	Lifted lead disconnects the velocity transducer signal from the monitor. The velocity signal is invalid causing spurious alarms. Problem to be corrected at the next available MODE 3 (forced outage) window	2	Jumper Log No. 89-018
2.	Temporarily lifted lead removes the automatic actuation of the Containment of the Containment Evacuation Alarm, generated by Source Range (NIS) instrumentation (High Flux at Shutdown)	2	Jumper Log No. 90-019

A. Facility Changes

1. Add Bearing Lift System for Low Pressure Turbine "C" Rotor Bearing No. 8
DCP M-16286 Rev. 2 (Unit 2)

The revised design of the spare low pressure turbine rotors required installation of a bearing lift system for the No. 8 bearing. It consists of a locally mounted pump with motor, filter, tubing and instrumentation. A local control console is mounted on the turbine pedestal and electric circuits in conduits connect to the existing turbine turning gear controls.

Safety Evaluation Summary

The bearing lift system: (1) is powered from non-vital power supplies; (2) does not interface with any Class 1E circuits; (3) is not connected with or immediately adjacent to any safety related or safe shutdown systems; (4) is not associated with any accident mitigating functions; (5) is not covered by any DCPP Technical Specifications; (6) does not compromise any existing 10 CFR 50 Appendix R reviews; (7) contains no radioactive materials; and (8) is Design Class II.

2. Remove Alarms from Reactor Protection System Cabinets' (Hagen Racks') Doors
DCP E-17759 Rev. 1 (Unit 1) and E-18759 Rev. 1 (Unit 2)

The reactor protection system (RPS) cabinets' doors were provided with interlocks to give an alarm indication in the control room when more than one protection set's cabinet doors were open. In addition, the test panels were similarly alarmed when opened to notify the control room of testing in progress. These alarms, status lights and their circuits were deleted by this design change. Energizing these circuits by opening the doors had been identified as a source of inductive pick-up in the RPS circuitry which caused spurious actuation of the RPS.

Safety Evaluation Summary

The deleted (non-class 1E) alarm circuits were independent of all RPS circuitry and logic. No accident detection or mitigation functions are affected by this change. These circuits were not governed by Technical Specifications.

The integrity of the protection set cabinets is not compromised by this modification. The control room is still notified when RPS testing is being performed via other existing alarms and through administrative controls required by procedures.

3. Provide Concrete Pad for the Mobile Radwaste Processing System
DCP M-29953 Rev. 10 (Units 1 and 2)

In conjunction with the new radwaste storage building provided under this DCP (reported in PG&E letter DCL-90-080), this design change package provided a new concrete slab just east of the fuel handling building. This pad provides an area for use of the mobile radwaste

A. Facility Changes (cont'd)

processing system (MRPS). Drainage/runoff from this outdoor pad drains to the plant's equipment drain receiver tanks.

Safety Evaluation Summary

As a replacement for an existing concrete pad on which the MRPS was used, the new pad has a larger capacity for collecting and draining any spills/leakage from the radwaste process as well as rainwater which collects on the pad. No change was made to radwaste processing or onsite or offsite doses as a result of the new pad configuration.

4. Abandon/Lay-up Radioactive Waste Concentrator Package
DCP M-31188 Rev. 9 (Units 1 and 2)

As reported in PG&E letter DCL-87-270, the radioactive waste concentrator was disconnected to allow for its conversion to a boric acid concentrator. Subsequent to the disconnection of the waste concentrator and prior to its full conversion to a boric acid concentrator, a decision was made to abandon the equipment in-place and to lay-up the waste concentrator for the present. Inasmuch as the equipment conversion originally planned had already been partially completed, the abandonment work consisted of placing the affected equipment and components in a safe condition and documenting the as-left condition.

Safety Evaluation Summary

The original waste concentrator equipment package was a Design Class II system and performed no safety-related functions. However, some concentrator equipment did interface with the safety-related component cooling water (CCW) system. Both the CCW system piping and the waste concentrator equipment it served had originally been seismically qualified as necessary to ensure the integrity of the CCW system.

It was determined, however, that as a consequence of the partially completed waste concentrator conversion effort, a portion of the CCW piping that served the waste concentrator package was no longer seismically qualified. To ensure that the design of the CCW system would not be compromised by the abandonment and lay-up of the waste concentrator, the isolation valves in the CCW supply and return headers to the waste concentrator package were sealed closed. By providing this positive isolation between the seismically-qualified and -unqualified portions of the CCW system, the availability and operability of the safety-related CCW system would not be adversely affected during or following a seismic event. The design of the header piping was also revised to include an appropriate piping code break at the sealed-closed header isolation valves to identify the end of the Design Class I CCW piping.

In addition to the changes made to the CCW header piping and isolation valves, all radioactive piping left open-ended as a consequence of the

A. Facility Changes (cont'd)

partially completed conversion of the waste concentrator was capped or plugged to preclude leakage of radioactively contaminated fluid to the equipment areas. Abandoned electrical circuits were terminated in junction boxes, open-ended conduits were either capped or routed to junction boxes, and heat tracing (installed as part of the conversion effort) was removed from uninsulated piping to ensure that no new fire hazards were created. All abandoned equipment and components were also carefully evaluated and appropriately modified to preclude any new seismically-induced system interaction concerns.

5. Revise Fire Detection and Suppression Control for the Emergency Diesel Generator Rooms
DCP M-33398 Rev. 1 (Unit 1) and M-34398 Rev. 1 (Unit 2)

This design change provides a new fire alarm control panel with back up battery power for each diesel generator room. The design of this system will allow electrical supervision for the fire detection system and provide for manual fire suppression carbon dioxide (CO_2) capability. Upon detection loop activation, the west roll up doors in the room will close and CO_2 will be discharged. Local, manual actuation is provided to meet surveillance testing requirements of the Technical Specifications. The west roll up doors in each diesel generator room must otherwise remain open to assure adequate cooling air for diesel generator operation.

Safety Evaluation Summary

The revised fire detection and suppression controls have been evaluated and installed to assure that a seismic event will not cause the roll up doors to close spuriously and compromise diesel generator operability. Circuits are provided with appropriate fire rated or engineering-evaluated protection and all fire barriers have been maintained. A failure of the Cardox system controls cannot result in closure of more than one set of doors.

6. Permit Replacement of Barton with Rosemount Pressure, Level and dP Transmitters
DCP J-33589 Rev. 0 (Unit 1) and J-34589 Rev. 0 (Unit 2)

This change provides for the replacement of pressure/level and differential pressure transmitters. Barton instruments can be replaced with those from Rosemount in various services throughout both Units as, for example, the containment emergency core cooling system (ECCS) recirculation sump narrow range level transmitters. The Barton devices are not field-repairable.

Safety Evaluation Summary

The replacement transmitters are seismically and environmentally qualified for the various plant areas in which they may be installed. The integrity of potential mounting locations (i.e., panels, etc.) will be maintained should replacements occur. The response and

A. Facility Changes (cont'd)

performance characteristics of the replacement instruments are suitable for and compatible with systems' requirements.

7. Install two, new, Liquid Radioactive Waste After-Filters
DCP M-35397 Rev. 1 (Units 1 and 2)

Two, new after-filters were installed in the liquid radioactive waste system. The filters are piped in parallel and located in the system as the last process equipment prior to radiation monitoring and discharge to the circulating water system. They are enclosed in existing, shielded vaults with provision for remote filter element changeout. The original after-filter will be used as one of three pre-filters upstream of the new filters.

Safety Evaluation Summary

This change affects only Design Class 2 systems and non-Class 1E circuits. It is installed per DCPP commitments to Regulatory Guide 1.143. This modification will not result in any change to liquid radwaste quantities and does not create or increase any hazards associated with systems interactions, flooding or radiological impact.

8. Convert the Waste Concentrates Holding Tank to a Second Boric Acid Concentrates Holding Tank
DCP M-39457 Rev. 0 (Units 1 and 2)

In conjunction with the abandonment/lay-up of the radioactive waste concentrator (see item above), the waste concentrates holding tank was converted to a second boric acid concentrates holding tank. Appropriate changes to piping, instrumentation and the auxiliary building control panel were also made.

Safety Evaluation Summary

All aspects of this design change (i.e., piping, instrumentation, controls, wiring and heat tracing) are Design Class II. No Technical Specifications relate to the affected equipment in this design change. The tank is located in a shielded vault suitable for its new service conditions/contents. Being no longer part of the liquid radwaste system (LRS), it is no longer associated with LRS inventory or flows. This tank and its use prior to and after its conversion does not affect the operability requirements of any systems in the plant Technical Specifications. This tank and its contents are not associated with any previously evaluated events in the safety analysis report (SAR).

9. Modify the Reactor Cavity Manipulator Crane Controls
DCP M-39515 Rev. 0 (Unit 1)

This design change performed the following modifications to the reactor cavity manipulator crane: 1) the Thyristor control system was replaced with a variable frequency control, 2) the existing pilot

A. Facility Changes (cont'd)

generators on the bridge and trolley drive motors were replaced with inching motors, and 3) a bridge camera/monitor position indication system was added.

The changes are used for assisting the crane operator in the exact positioning of the bridge and trolley, and to control the speed of the bridge, trolley and hoist motors.

Safety Evaluation Summary

The above modifications do not directly impact the handling of fuel assemblies or control rods.

Should the variable frequency control fail, the bridge and trolley would merely stop in place, not creating any potential for fuel damage. All existing fuel handling interlocks are unaffected by these changes. The crane remains seismically qualified for structural integrity. This design change adds an acceptably small amount of unqualified paint and no aluminum to existing containment inventories. No vital electrical loads or interfaces are changed.

Previous analyses of postulated fires, systems interactions, fuel handling accidents and loss-of-cooling accidents (LOCA) events still bound the potential contributions of the manipulator crane to these events.

10. Cut and Cap Spare Reactor Vessel CRDM Head Adaptors
DCP M-39666 Rev. 1 (Unit 1)

Four, spare reactor vessel control rod drive mechanism (CRDM) head adapters were replaced with full penetration butt welded pipe caps. The canopy seal weld for these head adapters had been leaking. This was reactor coolant pressure boundary (RCPB) work.

Safety Evaluation Summary

No unreviewed safety question was created and no Technical Specification change was required because:

* the materials used for the new closure are compatible with their normal and accident environments and operating conditions
* the new configuration does not change reactor coolant system (RCS) hydraulics
* all potential LOCA and missile effects resulting from a failure of this closure are bounded by existing analyses
* installation activities were controlled to prevent deleterious contamination of the RCS
* all appropriate Code analyses were performed to assure RCPB integrity
* CRDM cooling air distribution within the shroud has been assured by provision of a seismically mounted "dummy can" above each location
* Technical Specifications (TSs) requiring RCPB structural integrity,

A. Facility Changes (cont'd)

limits on RCPB leakage and an inservice inspection (ISI) program already bound the scope of this modification and do not require changing; moreover, the margins of safety of these TSs' bases are not reduced for the above reasons.

11. Install Digital Feedwater Control System
DCPs J-41250 Rev. 1, J-41373 Rev. 2 and E-41636 Rev. 0 (Unit 1) and J-42250 Rev. 1, J-42373 Rev. 1 and E-42636 Rev. 0 (Unit 2)

This modification replaces the existing analog-type steam generator water level control system with a fault-tolerant digital feedwater control system (DFWCS). The DFWCS utilizes a median signal selector (MSS) and was developed by the Westinghouse Owner's Group (WOG) as part of the Trip Reduction Program. The basic, functional design of the DFWCS is similar to the existing system at power levels above 20%. However, the DFWCS upgrade makes it possible to automatically control the steam generator level in the low power range (power level between 3% and 20%) by automatically regulating the corresponding bypass feedwater regulating valve. This reduces the required degree of manual manipulation of steam generator level controls at lower power levels and thereby reduces the potential for inadvertent reactor trips.

The hardware modifications consist of two seismically qualified cabinets in the main control room, a work station in the cable spreading room, temperature and pressure transmitters, a position transmitter for each of the main feedwater control valves, circuit isolators located inside the reactor protection system (RPS) racks, auto/manual control stations for the main feedwater control valves, bypass valves and main feedwater pumps. Also provided are interconnecting circuitry and conduits for the signals and a power supply for the system (including an uninterruptible power supply system). All necessary software to implement the DFWCS algorithms have also been provided.

Safety Evaluation Summary

The determination that the DFWCS modification does not involve an unreviewed safety question or a change to the Technic; Specifications is based on the following:

* Existing non-LOCA safety analysis previously evaluated in the safety analysis report (SAR) bound all potential effects/failures from the DFWCS. Other accident analyses are not affected.
* Design Class 1E isolators provide necessary isolation for interfaces with Design Class II systems.
* The new DFWCS cabinets have been seismically qualified and anchored for structural integrity to preclude any systems interactions concerns. The seismic qualification of the existing RPS racks and reactor vessel level instrumentation system (RVLIS) cabinets remains valid with the installation of DFWCS.

A. Facility Changes (cont'd)

* The MSS feature of the DFWCS meets the requirements of section 4.73 of IEEE-279 (1971) with respect to immunity from control/protection system interactions.
* The DFWCS modifications will have no effect on either the main or bypass feedwater control valve closure times. Neither will this change affect any of the safeguards systems (specifically, feedwater isolation) nor the response of any other design basis event-mitigation systems.

12. Lock Closed the Primary Water Makeup Supply to the CCW Surge Tank
DCP M-41291 Rev. 0 (Unit 1) and M-42291 Rev. 0 (Unit 2)

To prevent inadvertent addition of tritiated primary water storage tank (PWST) water to the component cooling water system, a chain lock was installed on makeup valve MU-1-1561 and MU-2-1561. The Design Class II manual valve is normally closed anyway. This change is in response to NRC I&E Bulletin 80-10.

Safety Evaluation Summary

The primary water storage tank (PWST), using the primary water transfer pumps (PWTPs), is only one of at least five sources of makeup water to the Design Class I, closed loop CCW system. The normal makeup supply is the transfer tank (150,000 gallon capacity) via the makeup water transfer pumps (MWTPs). The normal backup to this is the condensate storage tank using the MWTPs. These two sources are Design Class I. The locked PWST makeup valve will prevent contamination of the CCW system and does not materially reduce makeup options for CCW.

The added weight of the locking devise will not compromise the integrity of the Class I CCW makeup supplies. No systems' interactions are created by the chain/lock.

13. Disconnect Radiation Monitor RE-20 from the Radiation Monitoring System
DCP J-41463 Rev. 1 (Units 1 and 2)

RE-20 was provided to monitor the radiation level of the recirculating contents of the equipment drain receivers (EDRs). Prior to further processing in the liquid radwaste system (LRS), sampling of the recirculating fluid can still be used to monitor radionuclides in the EDRs. Other, downstream radiation monitors still exist in the LRS. RE-20 is an inline device which was left in place but with its wiring disconnected. It is Design Class II.

Safety Evaluation Summary

RE-20 does not provide any monitoring required by the Technical Specifications. Its elimination does not impact any quantity, processing, storage, sampling or discharge of radioactive waste. The radiation monitor is neither the source of nor required for detection

A. Facility Changes (cont'd)

or mitigation of any accidents or malfunctions described in the SAR. Other means (sampling and downstream radiation monitors) are provided to accomplish the intended function of this device.

14. Remove Moderate Energy Line Break Spray Guards providing protection for the Residual Heat Removal Pump Motors
DCP M-41490 Rev. 0 (Unit 1)

The 'moderate energy' seal water cooling piping below the residual heat removal (RHR) pump motors was originally thought to pose a hazard to the motor. Water spray shields, supported by the piping, were provided to protect the motors from the effects of a postulated moderate energy line break (MELB) in this piping. This DCP removes the shields. Any break in the seal water piping can only potentially affect its respective pump motor. The redundant RHR pump motor cannot be affected. This change is consistent with MELB evaluation criteria.

Safety Evaluation Summary

This removal of spray shields does not compromise the licensing or design basis for MELB at DCPP. Absence of the shields does not impact Technical Specification operability requirements or margins of safety for either the RHR or CCW systems. Seal water piping stresses remain within their Design Class I allowable. Other than for MELB mitigation, no credit was taken for the presence of the shields; therefore, no adverse impact on nuclear safety (or otherwise) results from their removal.

15. Disconnect the Plant Process Computer from Vital Power Supplies
DCP J-41527 Rev. 0 (Unit 1) and J-42527 Rev. 0 (Unit 2)

In conjunction with the replacement of the plant process computer (PPC), a replacement, uninterruptible power supply (UPS) was provided to power the computer. Formerly powered from the 480VAC and 125VDC Class 1E vital buses, the new PPC and its UPS are powered by the 480VAC and 125VDC non-vital buses, including a new 60-cell bank of non-vital batteries with a battery charger.

Safety Evaluation Summary

There is no requirement that the PPC be powered from a Class 1E, vital power source. The PPC is Design Class II. Apart from the new power sources not being seismically qualified, the UPS will still provide a highly reliable source of power for the PPC.

By removing the PPC load from the vital power buses, the operability of vital power systems required by the Technical Specifications can not and will not be adversely affected.

The areas in which the UPS components are installed (in the auxiliary building and turbine buildings) have been evaluated to assure the

A. Facility Changes (cont'd)

following issues have been adequately addressed: electrical equipment-generated heat loads, potential hydrogen generation in the battery room, added structural loading in important-to-safety buildings due to equipment weights and seismically-induced systems interactions.

16. Install Air Conditioning System for the Plant Process Computer Room
DCP J-41528 Rev. 1 (Unit 1)

The plant process computer (PPC) is located within the control room enclosure but in a room separated from the main control room. The new PPC introduces additional heat load in the computer room; a supplemental, Design Class II air conditioning system is provided in the room to remove the PPC heat load. The evaporator (cooling coil) units are in the computer room and the condenser units are mounted on the roof of the control room enclosure. Freon tubing, cooling coil condensate drain piping and electrical conduits penetrate the enclosure boundary for the evaporator units. Non-vital power supplies this air conditioning system.

Safety Evaluation Summary

The design classification of the air conditioning system is compatible with its functional requirement of providing cooling for the Design Class II PPC. The determination that no unreviewed safety question (USQ) or Technical Specification change is created by this design change is based on having adequately addressed the following issues:

* the control room remains habitable with an assumed release of freon into the control room air space
* the control room enclosure boundary integrity is maintained with sealed, engineering-evaluated penetrations; the basis for Technical Specifications for control room ventilation and fire barrier penetrations are not compromised
* seismically-induced systems interactions are precluded by design
* wall and roof loads from the new equipment are accounted for

17. Disconnect and Remove Power Range Neutron Flux Recorders
DCP J-41569 Rev. 1 (Unit 1) and J-42569 Rev. 2 (Unit 2)

As part of a major replacement of main control room recorders, the power range neutron flux recorders were disconnected and removed. There were four of these 2-pen recorders to record top and bottom channel core flux in the power range.

Safety Evaluation Summary

The seismic and fire protection integrity of the main control boards was maintained in conjunction with this work. Electrical separation and systems interacting requirements were similarly, adequately addressed in the work.

A. Facility Changes (cont'd)

These recorders are: 1) not modeled nor relied upon in any plant accident analyses, 2) not referenced in the Technical Specifications, 3) not required for accident prevention, mitigation or (dose) evaluation (plant transients can be assessed using other, existing power range recordings) and 4) not required by any emergency response procedures.

18. Install Quick Opening Fuel Transfer Tube Closure Assembly
DCP M-41650 Rev. 0 (Unit 1) and M-42650 Rev. 0 (Unit 2)

The inside-containment-end of the fuel transfer tube was originally provided with a blind flange. The blind flange was removable to provide for fuel transfer operations but otherwise remained installed. It has been replaced with a quick opening transfer tube closure (QOTTC). This change facilitates preparation for fuel transfer operations and restoration of containment integrity following such operations. The change is consistent with as low as reasonably attainable (ALARA) principles. The QOTTC forms part of the passive, Containment integrity boundary.

Safety Evaluation Summary

The QOTTC is Design Class I, Seismic Category I and designed per appropriate code requirements. It is fabricated of materials consistent with their operating (including accident) environments. As with the former blind flange, it contains double, O-ring seals for assuring closure integrity by periodic testing. The transfer tube and containment structural integrity have been verified for this new attachment to the tube.

Fuel handling operations are not compromised by installation of the QOTTC. Technical Specification requirements and margins of safety related to containment integrity, containment structural integrity, refueling operations and radioactive effluents are not compromised by the installation of the QOTTC.

19. Replace the Sequence of Events Recorder Subsystem of the Main Annunciator System
DCPs E-41723 Rev. 0 (Unit 1) and E-42352 Rev. 0 and J-42534 Rev. 0 Unit 2)

The main annunciator system sequence of events recorder (SER) has been replaced in each unit. This work included equipment in the cable spreading room and the main control room (alarm printer and a new microprocessor-driven cathode ray tube (CRT)).

Safety Evaluation Summary

The SER is Design Class II. However, the annunciator cabinets and printer have been qualified to function during and after seismic events. The CRT portion of the system on the control operator's console, though not required to function following a seismic event,

A. Facility Changes (cont'd)

has been seismically qualified so as to not create any systems interactions in the control room. The new SER is powered by the same Class 1E power as formerly with only a slight (and acceptable) increase in load. Suitable circuit breaker protection for the Class 1E source is existing. The SER is isolated from Class 1E alarm signal sources. The replacement of this system does not affect any fire hazards analysis. The change in heat load from the equipment has an inconsequential impact on heating ventilation and air conditioning (HVAC) systems.

20. Reconfigure/Upgrade Seismic Trip System
DCPs J-41746 Rev. 1 and J-43064 Rev. 4 (Unit 1) and J-42746 Rev. 1 and J-44064 Rev. 0 (Unit 2)

The purpose of this DCP is to remove the relay-based logic of the existing seismic reactor trip system and replace it with a system that incorporates voting logic into the plant solid state protection system (SSPS). This modification is being implemented because of periodic testing difficulties experienced by the plant (and the resulting false Reactor trips) with the existing relay based system. The new system upgrades the existing seismic sensor packages to Design Class I (because of their interface with the Design Class I SSPS) and routes the signals from these seismic sensors to the SSPS cabinets via three "Bypass/Isolate" switches, each mounted in a separate, seismically supported panel. The new system provides ample annunciation features to alert operations/maintenance personnel of the existing status of the system. The new conduits are seismically supported with required separation being maintained between SSPS channels I, II and III. The modification within the SSPS includes addition of new logic cards and new relays.

Safety Evaluation Summary

Westinghouse provided the safety evaluations for modifications within the SSPS. It concluded that the additional head load of 7.3 watts has a negligible impact on SSPS normal operating temperatures. Also, the additional weight of only 2.76 pounds does not affect the seismic qualification of the existing SSPS cabinets. Finally, the response time of the seismic reactor trip function decreases due to this modification. Since the DCPP Technical Specifications do not specify a required response, the plant licensing basis is not impacted. In addition, there is no impact on the response time of other plant protection systems.

For modifications outside the SSPS, the new components and panels are seismically supported and the existing seismic sensors have been requalified to Design Class I. Signals going to the SSPS are routed in seismically supported conduits with adequate separation being maintained between Channels I, II and III. Design Class I signals interfacing with the annunciators, computers and light box (Design Class II components) have been adequately isolated. Issues such as human factors have been adequately addressed. Since no credit is

A. Facility Changes (cont'd)

taken for functioning of the seismic trip in any of the SAR-analyzed accidents (see Technical Specification Bases 2.2), this modification has no effect on the margin of safety as defined in any Technical Specification.

21. Install Cathode Ray Tubes and Personal Computers in the Control Console in the Main Control Room
DCPs J-42533 Rev. 1 and J-42534 Rev. 0 (Unit 2)

As part of the replacement of the plant process computer, modifications were made to the main control boards in the control room; specifically, three, new CRTs and personal computers were installed in the operators' control consoles.

Safety Evaluation Summary

The information provided to plant operators by this Design Class II video data is not for detection or required mitigation of any accident.

The new equipment is: 1) mounted such that the required seismic integrity of the control consoles is maintained, 2) physically separated from the Class 1E portions of the control console so no new systems interactions are created, 3) built of materials which are compatible with a habitable control room environment and 4) accounted for appropriately in the combustibles loading in the control room.

22. Raise the Condensate Storage Tank Low-Low Level Alarm Setpoint
DCPs J-43013 Rev. 0 (Unit 1) and J-44013 Rev. 0 (Unit 2)

The setpoint for the condensate storage tank (CST) low-low level alarm switch, LS-464, was raised from 30 inches to 36 inches above the centerline of the 12 inch auxiliary feedwater (AFW) nozzle (N-1). The level switch is a Class IB, type A, category I device, which provides a low-low level alarm in the control room and performs no other function.

Safety Evaluation Summary

The alarm setpoint was raised to meet a PG&E commitment to the NRC. The new setpoint assures at least 20 minutes for operator action to realign the AFW water supply before potentially losing the CST supply at an assumed flow corresponding to the largest AFW pump operating. The setpoint accounts for prevention of vortex formation at these low tank levels. This change does not impact the Technical Specification requirements for minimum CST water volume.

A. Facility Changes (cont'd)

23. Install Fiberoptic Inspection Ports in the Main Steam Isolation Valves
DCP P-43028 Rev. 0 (Unit 1) and P-44028 Rev. 0 (Unit 2)

A new inspection port connection was provided for each main steam isolation valve (MSIV) upstream of the valve disk. The connection consists of short, welded pipe nipples with a normally closed gate valve and threaded cap. This will allow video/fiberoptic inspection of the MSIV internals during outages without valve disassembly.

Safety Evaluation Summary

Within the main steam, containment penetration boundary, these connections meet all appropriate design, material, construction and testing requirements for containment penetration piping. Seismic/stress calculations have documented their integrity. No new seismically induced systems interactions are created by their addition. Previously evaluated accidents and malfunctions in the SAR envelop all potential failures of these new connections. Appropriate ALARA considerations and Code and ISI requirements have been applied.

24. Install Nitrogen Supply System for the I&C Shop Calibration Lab
DCP A-43141 Rev. 0 (Units 1 and 2)

The nitrogen (N_2) supply for the I&C shop calibration lab was augmented to include liquid N_2 storage tanks, a booster compressor, high pressure storage bottles for N_2 gas storage and necessary controls, all located at the north end of the I&C shop. Motive power for the booster compressor is compressed air from the Design Class II, compressed air system.

Safety Evaluation Summary

The storage containers operate at a pressure greater than 100 psig and are, therefore, a potential source of missile damage to important-to-safety plant features. Missile generation likelihood is remote due to conservative design, overpressure protection, physical restraint of the bottles, remoteness from important-to-safety targets and protection from vehicular traffic. Habitability of the control room has been re-verified with the addition of this bulk N_2 storage, a potential asphyxiant to personnel.

25. Install new Reactor Vessel Stud Tensioner Hoists
DCP M-43298 Rev. 0 (Unit 1) and M-44298 Rev. 0 (Unit 2)

To expedite Reactor Vessel (RV) head stud de-tensioning and re-tensioning during refueling operations, a total of six electrically operated hoists have been installed in each Unit. They are installed

A. Facility Changes (cont'd)

on the circular monorail supported from the CRDM seismic support assembly. Each hoist is rated at two tons and is provided with a motorized trolley. The hoists are used to lift and move the stud tensioners and studs.

Safety Evaluation Summary

The following points assure that the issues addressed by 10 CFR 50.59 do not result in a USQ or require a change to the Technical Specifications:

1) As installed on the monorail above the RV flange, the hoists are seismically qualified to remain on the rail; they cannot fall and potentially compromise the RCPB.
2) The hoists and trolleys are powered from non-vital, 480 VAC motor control centers (MCCs) and, additionally, are in use only during outages.
3) The small, additional amount of insulated electrical cable added to power the hoists does not adversely affect the 10 CFR 50 Appendix R analysis for this fire zone.
4) The appropriate requirements for heavy loads-handling hoists (including load tests) are applied to minimize the risk of a load drop while these hoists are in use.
5) The added weight of the hoists to the RV head assembly has been documented as acceptable regarding: a) RV head lifting rig integrity and b) RV head assembly (seismic) integrity.
6) The resulting addition of aluminum and paint inside Containment have been accounted for in the inventory tabulations for these commodities; no accident analysis assumptions, inputs or results are changed as a result.
7) No plant Technical Specifications relate to these modifications.

26. Alter and Document the Configuration of the Containment Recirculation Sump
DCPs C-43642 Rev. 0 and C-43762 Rev. 1 (Unit 1) and C-44642 Rev. 0 and C-46080 Rev. 0 (Unit 2)

Design changes were issued for each Unit to alter and document the configuration of the containment recirculation sumps. The physical changes include: 1) providing a 3/16" opening mesh screen over the inner (lower) grating assembly in the Unit 1 sump, 2) restoring support for small areas of screen covering the upper grating assembly in the Unit 1 sump and 3) covering openings in the upper and lower grating assemblies larger than 3/16" in size in both Units.

Safety Evaluation Summary

The containment recirculation sump is a passive structural component required for the longer-term mitigation of loss-of-coolant accident (LOCA) events. The final configuration documented as a result of these design changes satisfies the following:

A. Facility Changes (cont'd)

1) Seismic integrity of the sump is maintained.
2) Adequate open screen area remains available assuming maximum, credible blockage by insulation debris or failed coatings inside containment.
3) Adequate RHR pump net positive suction head (NPSH) is maintained without air entrainment or vortexing.
4) Postulated debris greater than 3/16" in size is restricted from entering the RHR system.
5) Operability of ECCS, the containment spray system and post-LOCA Sump water level instrumentation, all required by Technical Specifications, is not compromised.
6) Physical separation of the two RHR pump suction lines is maintained.
7) The screened sump structure is not subject to damage from any high energy line breaks or missiles resulting from/in LOCAs.
8) All materials used are compatible with the normal and accident environments land conditions.

27. Disconnect Air Supply Through Containment Penetration 80 and Cap the Penetration
DCN 0-EM-1106 Rev. 0 (Units 1 and 2)

This change disconnected an Instrument Class 1A instrument air supply passing through containment penetration #80 and which was no longer required. Containment isolation valve FCV-682 was removed from this air supply line and penetration #80 was capped at both ends. The Instrument Class 1A endpoints served by the air supply were also capped.

Safety Evaluation Summary

The integrity of penetration #80 (and, hence, containment integrity) was maintained by installation of flareless fitting caps on both ends of the penetration tubing. Similarly, integrity of the air tubing endpoints was re-established with caps. FCV-682 was removed as a load from vital electrical power.

28. Remove Radwaste Solidification/Drumming/Storage Facility from Fuel Handling Building
DCN DC2-EM-1257 Rev. 0 (Units 1 and 2)

This change removed the solid radwaste facility from the fuel handling building in preparation for use of the mobile radwaste processing system and the new radwaste storage building.

Safety Evaluation Summary

The requirements of Technical Specification 3.11.3 for processing of solid radioactive waste are being met by the provision of a mobile radwaste processing system. The original permanently-installed facility was removed by this design change. These changes to the

A. Facility Changes (cont'd)

solid radwaste treatment system have been previously reported under Technical Specification 6.15.

29. Replace Containment Electrical Penetration Assemblies 21E and 39E DCNs 1-EC-19688 Rev. 4 and 1-EE-19688 Rev. 1 (Unit 1) and 2-EC-20688 Rev. 1 and 2-EE-20688 Rev. 0 (Unit 2)

New, Conax, 'feedthru' type electrical penetration assembly modules were installed in existing, empty sleeves in both Units at containment penetration locations 21E and 39E. These new electrical penetration assemblies were added to support the addition of the new post-accident neutron flux monitors previously reported in PG&E letter DCL-86-300. The new electrical penetration assemblies are Design Class I, Class 1E, and are both environmentally and seismically qualified in accordance with applicable DCPP environmental and seismic qualification criteria.

Safety Evaluation Summary

The new containment electrical penetration assemblies were installed in a manner similar to the existing G.E. 'canister' type electrical penetration assemblies. As such they are designed to ensure primary containment integrity. To further ensure that primary containment integrity had not been compromised by their installation, appropriate post-installation testing of the installed penetration assemblies was performed. Inasmuch as the electrical design of the new penetration assemblies is consistent with that of the new post-accident neutron flux monitoring equipment, they will not compromise or otherwise adversely affect the capability of the flux monitors to meet the requirements of Regulatory Guide 1.97.

B. Procedure Changes

1. Temporary Change to Procedure OP L-4 For Unit 1

A 5 degree reduction in Tavg for End Of Life (EOL), Cycle 3 operation for Unit 1

Safety Evaluation Summary

The effect of a 5 deg F Tavg/Power coastdown for Diablo Canyon Unit 1 Cycle 3 has been evaluated by Westinghouse. The potential effect of the proposed coastdown on the FSAR Update accident analysis results for each of the accident analyses was evaluated and it was shown in all cases that the effect of the change did not result in exceeding any design or regulatory limit. Therefore, it can be concluded that the proposed EOL 5 deg F Tavg/Power coastdown for Diablo Canyon Unit 1 is acceptable from the standpoint of FSAR Update accident analyses.

A. Facility Changes (cont'd)

C. Procedures (Test)

1. Startup Test Procedure SU 23.3A7 Rev. 0, Unit 2

This procedure is for operational testing of auxiliary building supply and exhaust fan inlet damper vanes to obtain fan flow switch d/p data and to verify that replacement controllers for the inlet vanes perform per design.

Safety Evaluation Summary

The presently approved conditions for the Unit 2 auxiliary building ventilation system will not be affected in any manner by this procedure. Appropriate measures are written into the procedure to restore the supply ventilation prior to exceeding a Technical Specification limit.

2. Startup Test Procedure SU35.6 A2, Rev. 0, Unit 1

This procedure functionally tests the containment ventilation isolation (CVI) system after implementation of a change which provides an alternate actuation circuit and 120 VAC power source.

Safety Evaluation Summary

This procedure verifies the operability of the CVI system after the implementation of an approved design change.

3. Temporary Procedure TB-8934 Rev. 0, Unit 1

This is a one time test of the steam dump turbine trip controller actuation which was changed from actuation by SSPS slave relays to actuation by reactor trip relays RTK A/B. The actuation is the result of the implementation of an approved design change.

Safety Evaluation Summary

This test is performed in MODE 5 when the steam dump system is not required. The control rod motor generator (MG) sets are checked to be shutdown, so no control rods can be withdrawn when the reactor breakers are closed. Steam dump controls are placed in blocking position so no dump valves open during the test.

4. Temporary Procedure TB-8929 Rev.0 Unit 1

A temporary (Test) procedure to verify the post-replacement wiring and correct operation of SSPS slave relay test switches.

C. Procedures (Test) (cont'd)

Safety Evaluation Summary

The replacement switch circuits provide only a blocking signal to the steam dump system and can not open any valve. The steam dump system is not required in MODES 5-6 when this test will be done.

5. Temporary Procedure TB-8905, Rev. 0, Units 1 and 2

This procedure is used to determine the margin of voltage and current in the MSIV solenoid continuity test circuit. Pickup voltage and current is measured for each MSIV solenoid to determine if the dropping resistors in the continuity test circuit are properly sized.

Safety Evaluation Summary

This test is a one time test and will be performed in MODES 3, 4, 5 or 6 and not during normal power operation. The test does not permanently modify circuits of the MSIV trip solenoids.

6. Temporary procedure, TB-8906, Rev. 0, Units 1 and 2

This procedure is used to determine the margin of voltage and current in the anti-motoring and sequential trip relay test circuits. The data obtained will be used to determine if the dropping resistors in the continuity test circuits are properly sized.

Safety Evaluation Summary

No permanent modification to the plant circuitry is made. The test is only performed in MODES 3, 4, 5, or 6 when the anti-motoring and sequential trip relays are not required.

7. Temporary Procedure, TB-8909, Rev. 0, Units 1 and 2

This procedure is used to measure the pickup voltage and current for the turbine trip solenoids. The data obtained will be used to determine if the dropping resistors in the continuity test circuits are sized right.

Safety Evaluation Summary

No permanent modification is made to the plant circuitry. The test is performed in MODES 3, 4, 5, or 6 only and not during power operation.

8. Surveillance Test Procedure STP V-21 Rev. 0, Units 1 & 2

This procedure provides a leak test of spray additive tank outlet line check valves 8998A and 8998B.

C. Procedures (Test) (cont'd)

Safety Evaluation Summary

The containment spray system is not required to be operable in the MODES (MODES 5 and 6) in which this procedure will be performed. In accordance with Technical Specification 3.1.2.5, a boric acid storage system (e.g., boric acid strorage tank (BAST)) or raw water storage tank (RWST) is required to be operable in MODES 5 & 6 for operations involving core alterations or positive reactivity changes. This procedure will not render inoperable the BAST, the RWST, or their associated injection systems. Core alterations or positive reactivity changes could quickly and easily be stopped if the operability of the RWST was in question.

9. Surveillance Test Procedure STP M-112 Rev.0, Unit 1

This procedure pressurizes the boron injection tank (BIT) discharge piping to a nominal reactor coolant system pressure in order that an external inspection of the piping may be accomplished.

Safety Evaluation Summary

All of the components of test equipment will have a service rating higher than the piping to which it will be attached. The hydro pump rig will be tested for integrity at 2215 psig (normal test pressure) prior to being placed in service. The hydro pump discharge includes a check valve installed close to the boron injection tank discharge piping test tap valve. The test apparatus will be constantly attended while attached to the test tap and constant phone communications will be maintained with the control room. The test equipment will be used such that there is no impact on the ability of the BIT system to function at any time.

D. Jumpers/Lifted Leads

1. Lifted Lead On RCP-2-1 Motor Frame Channel A Velocity-X YT-455A Transducer Jumper Log No. 89-018 Unit 2

This lifted lead disconnects the velocity transducer signal from the monitor. The velocity transducer signal is invalid causing spurious alarms. Problem to be corrected at the next available MODE 3 (forced outage) window.

Safety Evaluation Summary

The lead was lifted to disconnect the input signal from the transducer as it has been determined that the signal is no longer valid and has been causing spurious alarms in the control room. The functional requirements of the reactor coolant pump vibration monitoring system will continue to be met by the remaining five

C. Procedures (Test) (cont'd)

Safety Evaluation Summary

The containment spray system is not required to be operable in the MODES (MODES 5 and 6) in which this procedure will be performed. In accordance with Technical Specification 3.1.2.5, a boric acid storage system (e.g., boric acid strorage tank (BAST)) or raw water storage tank (RWST) is required to be operable in MODES 5 & 6 for operations involving core alterations or positive reactivity changes. This procedure will not render inoperable the BAST, the RWST, or their associated injection systems. Core alterations or positive reactivity changes could quickly and easily be stopped if the operability of the RWST was in question.

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This procedure pressurizes the boron injection tank (BIT) discharge piping to a nominal reactor coolant system pressure in order that an external inspection of the piping may be accomplished.

Safety Evaluation Summary

All of the components of test equipment will have a service rating higher than the piping to which it will be attached. The hydro pump rig will be tested for integrity at 2215 psig (normal test pressure) prior to being placed in service. The hydro pump discharge includes a check valve installed close to the boron injection tank discharge piping test tap valve. The test apparatus will be constantly attended while attached to the test tap and constant phone communications will be maintained with the control room. The test equipment will be used such that there is no impact on the ability of the BIT system to function at any time.

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1. Lifted Lead On RCP-2-1 Motor Frame Channel A Velocity-X YI-455A Transducer Jumper Log No. 89-018 Unit 2

This lifted lead disconnects the velocity transducer signal from the monitor. The velocity transducer signal is invalid causing spurious alarms. Problem to be corrected at the next available MODE 3 (forced outage) window.

Safety Evaluation Summary

The lead was lifted to disconnect the input signal from the transducer as it has been determined that the signal is no longer valid and has been causing spurious alarms in the control room. The functional requirements of the reactor coolant pump vibration monitoring system will continue to be met by the remaining five

C. Procedures (Test) (cont'd)

monitors on the pump. The vibration monitoring system is Design Class II and a malfunction is not postulated or evaluated by the FSAR Update.

2. Lifted Lead on Auto-Initiated Containment Evacuation Alarm Jumper Log No. 90-019 Unit 2

This lifted lead removes the automatic actuation of the Containment Evacuation Alarm generated by the nuclear instrumentation systems (NIS) source range instrumentation (high flux at shutdown).

Safety Evaluation Summary

The automatic actuation of containment evacuation alarm described in the FSAR Update Section 14.1.4.1 is defeated by lifting this lead. However, manual actuation of the alarm from the main control board, CC1, is not affected. The automatic feature was removed to eliminate spurious alarms. No credit is taken for this automatic alarm actuation in relation to fuel handling accidents as described in the FSAR Update. The lead is to be reconnected prior to leaving MODE 4.

Westinghouse Electric Corporation Energy Systems

March 20, 1991
NS-OPLS-91-122

Document Control Desk
US Nuclear Regulatory Commission
Washington, DC 20555

Attention: Dr. Thomas Murley, Director

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: WCAP-11953, "Safety Evaluation Supporting a More Negative EOL Moderator Temperature Coefficient for the Joseph M. Farley Nuclear Power Plant Units 1 and 2".

Dear Dr. Murley:

Affidavit CAW-89-003 signed by the owner of the proprietary information, Westinghouse Electric Corporation, sets forth the basis on which the information contained in WCAP-11953 may be withheld from public disclosure by the Commission.

It has come to our attention that the date on the Affidavit is not correct. Westinghouse agrees that a typographical error exists on the document and does hereby certify that the affidavit is valid and was notarized January 10, 1989. Attached is a notarized copy of the Notarial Register in support of this.

This letter shall be used to confirm the validity of Affidavit CAW-89-003. Correspondence related to CAW-89-003 should be addressed to the undersigned.

Very truly yours,

R. P. DiPiazza, Manager
Operating Plant Licensing Support

Enclosures

cc: M. P. Siemien, Esq.
Office of the General Counsel, NRC

S. T. Hoffman, APCo Project Manager

0056J:RJS/032091

TOIC

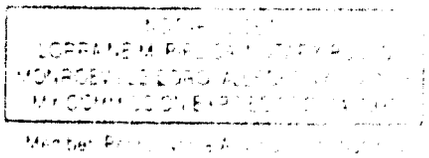
CERTIFICATE

COMMONWEALTH OF PENNSYLVANIA)
) SS:
COUNTY OF ALLEGHENY)

I certify that the attached copy of two pages from my Official Notarial Register, listing my notarial acts from November 9, 1988 to and including January 31, 1989, is a true, correct and complete copy of the original Notarial Register.

In witness whereof, I hereunto set my hand and official seal.

Loraine M. Kolcun
Notary Public



March 15, 1991

NOTARIAL REGISTER

DATE OF NOTARIAL ACT Mo	Day	Year	CHAR OF ACT	EXECUTED BY	TO	EXECUTED
11	12	88	1			Eric G. Davidson

DATE OF INSTRUMENT Mo	Day	Year	NOTARY FEE PAID	REMARKS
				(4)
				Title Transfer
				CAW 88 097 - 2 other
				CAW 88 127
				W.B.
				Power of Attorney
				Power of Att. &
				P.E. Application
				Mont.
				(5)
				MV-38
				Title
				CAW 89 003
				CAW-88-132

CAW-89-003

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

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COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared Robert A. Wiesemann, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Corporation ("Westinghouse") and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

Robert A. Wiesemann, Manager
Regulatory and Legislative Affairs

Sworn to and subscribed
before me this ___ day
of _____, 1988.

Notary Public

NOTARIAL SEAL
LORRAINE M. PALCA, NOTARY PUBLIC
MONROEVILLE BORO, ALLEGHENY COUNTY
MY COMMISSION EXPIRES DEC. 14, 1991

Member, Pennsylvania Association of Notaries

(1) I am Manager, Regulatory and Legislative Affairs, in the Nuclear and Advanced Technology Division, of the Westinghouse Electric Corporation and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rulemaking proceedings, and am authorized to apply for its withholding on behalf of the Westinghouse Energy Systems, Nuclear Fuel, and Power Generation Business Units.

(2) I am making this Affidavit in conformance with the provisions of 10CFR Section 2.790 of the Commission's regulations and in conjunction with the Westinghouse application for withholding accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by the Westinghouse Energy Systems, Nuclear Fuel, and Power Generation Business Units in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.790 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

(g) It is not the property of Westinghouse, but must be treated as proprietary by Westinghouse according to agreements with the owner.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information which is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10CFR Section 2.790, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in "Safety Evaluaiton Supporting a More Negative EOL Moderator Temperature Coefficient Technical Specification for the Joseph M. Farley Nuclear Plant Units 1 and 2," WCAP-11953 (Proprietary), for J. M. Farley Nuclear Plant, Units 1 and 2, being transmitted by the Alabama Power Company (APCo) letter and Application for Withholding Proprietary Information from Public Disclosure, W. G. Hairston, III, APCo, to the Attention of Thomas Murley (NRC), Document Control Desk, January, 1989. The proprietary information as submitted for use by Alabama Power Company for the J. M. Farley Units 1 and 2 is expected to be applicable in other licensee submittals in response to certain NRC requirements for justification of a more negative End of Life Moderator Temperature Coefficient.

This information is part or that which will enable Westinghouse to:

(a) Provide documentation of the analyses, methods and conservatisms for reaching a conclusion relative to reducing EOL MFC Limiting Condition for Operation (LCO) and Surveillance Requirements (SR).

(b) Identify benefits for the suspension of the Surveillance Requirements below 100 ppm Boron Concentration.

(c) Establish the effects on the present safety analysis with the More Negative MTC conditions.

(d) Establish temperature and pressure affects and RCCA Insertion Limits for EOL MTC.

(e) Assist customer to obtain NRC approval.

Further this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of similar information to its customers for purposes of meeting NRC requirements for licensing documentation.

(b) Westinghouse can sell support and defense of the analyses to its customers in the licensing process.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar analytical documentation and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended for developing the analytical methods and approach.

Further the deponent sayeth not.

71-5797

Department of Energy
Washington, DC 20585

MAR 22 1991

Mr. Charles E. MacDonald
Chief, Transportation Branch
Division of Safeguards and Transportation, NMSS
Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. MacDonald:

This letter notifies you that DOE wishes to renew NRC Certificate of Compliance USA/5797/B()F which expires April 31, 1991. DOE plans to submit a revised SARP in Reg Guide 7.9 format along with a renewed DOE Certificate of Compliance and Safety Evaluation Report (SER) issued by EH-321 in support of this application for renewal. These are not presently ready but EH-321 desires and expects to present them before the NRC's Certificate expires. If the revised SARP and DOE's Certificate of Compliance and SER are not finished before April 31, 1991, EH-321 plans to cancel DOE's "timely renewal" on this package and allow the NRC Certificate of Compliance to expire.

If you have any comments or questions, please contact Frank Punch (353-6020) or Bob Towell (353-3954) of my staff.

Sincerely,

Jacques Read

Jacques Read
Acting Director
Division of Quality Verification
and Transportation Safety

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Log # TXX-91120
File # 10013 w/o enc.
Ref. # 10CFR50.54(q)
10CFR50 App. E

William J. Cahill, Jr.
Executive Vice President

March 22, 1991

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES)
DOCKET NOS. 50-445 AND 50-446
TRANSMITTAL OF REVISED EMERGENCY PLAN PROCEDURES

Gentlemen:

Enclosed is (1) copy of each of the Emergency Plan Procedures (EPP) (Controlled Copy Number 754) listed on the attachment.

If you have any questions regarding these changes, please contact Mr. Greg Bell at (214) 812-4373.

Sincerely,

William J. Cahill, Jr.

William J. Cahill, Jr.

By: Roger D. Walker
Roger D. Walker
Manager of Nuclear Licensing

GLB/grp
Attachment
Enclosures

c - Mr. R. D. Martin, Region IV (2 copies of enclosures; Control Numbers 754A and 754B)
Resident Inspectors, CPSES (3) (Information Only Copies of enclosures)
Mr. J. Clifford, NRR (1 Information Only Copy of enclosures)

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EMERGENCY PLAN PROCEDURES, FORMS, AND INDICES

Procedure Index, Revision 128

Procedure Index, Revision 129

Procedure Index, Revision 130

Procedure EPP-304, Revision 11, "Protective Action Recommendations"

Form Index, Revision 28

COMANCHE PEAK STEAM ELECTRIC STATION

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*EPP-101 (eff. 05/05/89)	7	ADMINISTRATIVE CONTROL OF EMERGENCY PLAN, AND ASSOCIATED PROCEDURES AND FORMS	Proc.Retired (replaced by EPP-900)
*EPP-102	4	REVIEW, UPDATE AND CONTROL OF THE EMERGENCY PLAN	Proc.Retired
*EPP-103	1	DUTIES OF THE EMERGENCY COORDINATOR	Proc.Retired
*EPP-104 (eff. 01/30/89)	5	EMERGENCY PREPAREDNESS DRILLS AND EXERCISES	Proc.Retired (replaced by EPP-902 R-0)
*EPP-105	1	NOTIFICATION OF OFFSITE OFFICIALS PENDING FLOOD OR FAILURE OF THE SQUAW CREEK RESERVOIR DAM	Proc.Retired
*EPP-106	1	SURVEILLANCE OF EMERGENCY EQUIPMENT	Proc.Retired
*EPP-107 (eff. 04/05/89)	7	MAINTENANCE AND INVENTORY OF EMERGENCY EQUIPMENT AND SUPPLIES	Proc.Retired (replaced by EPP-901 R-0)
*EPP-108 (eff. 07/20/89)	6	CONTROL OF DESIGNATED EMERGENCY USE VEHICLES USED FOR ROUTINE OPERATIONS	Proc.Retired (replaced by EPP-905 R-0)
*EPP-109 (eff. 05/11/90)	8	DUTIES AND RESPONSIBILITIES OF THE EMERGENCY COORDINATOR/RECOVERY MANAGER	09/18/91
*EPP-110	1	CORPORATE SUPPORT	Proc.Retired
*EPP-111	1	OFFSITE SUPPORT AND ASSISTANCE	Proc.Retired
*EPP-112 (eff. 05/11/90)	7	DUTIES OF CONTROL ROOM PERSONNEL DURING EMERGENCIES	09/16/91
*EPP-113	1	CORPORATE SUPPORT	Proc.Retired
*EPP-114	1	OFFSITE SUPPORT AND ASSISTANCE	Proc.Retired

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*EPP-115 (eff. 05/05/89)	5	RELEASE OF EMERGENCY-RELATED INFORMATION TO THE PUBLIC	Proc.Retired (replaced by EPP-207 R-5)
*EPP-116 (eff. 05/11/90)	4	EMERGENCY REPAIR AND DAMAGE CONTROL AND IMMEDIATE ENTRIES	09/24/91
*EPP-117 (eff. 03/30/90)	5	ADMINISTRATIVE CONTROLS FOR SAMPLING UNDER POST ACCIDENT CONDITIONS	02/04/92
*EPP-118 (eff. 03/16/90)	4	ALERT AND NOTIFICATION SYSTEM SURVEILLANCE & REPORTS	Proc.Retired (replaced by EPP-907 R-0)
*EPP-119 (eff. 03/31/89)	1	SOUND POWERED TELEPHONE SYSTEM AND EMERGENCY EVACUATION ALARM SYSTEM PERIODIC TESTING	Proc.Retired
*EPP-121 (eff. 05/11/90)	6	REENTRY, RECOVERY AND CLOSEOUT	09/14/91
*EPP-201 (eff. 12/22/89)	7	ASSESSMENT OF EMERGENCY ACTION LEVELS, EMERGENCY CLASSIFICATION AND PLAN ACTIVATION	09/17/91
*EPP-202 (eff. 01/11/91)	4	EMERGENCY COMMUNICATIONS SYSTEMS AND EQUIPMENT	08/26/91
*EPP-203 (eff. 05/26/89)	9	NOTIFICATIONS	10/10/91
*EPP-204 (eff. 11/09/90)	8	ACTIVATION AND OPERATION OF THE TECHNICAL SUPPORT CENTER (TSC)	08/08/91
*EPP-205 (eff. 09/28/90)	7	ACTIVATION AND OPERATION OF THE OPERATIONS SUPPORT CENTER (OSC)	09/14/91
*EPP-206 (eff. 11/09/90)	8	ACTIVATION AND OPERATION OF THE EMERGENCY OPERATIONS FACILITY (EOF)	09/16/91
*EPP-207 (eff. 07/20/89)	6	ACTIVATION AND OPERATION OF THE NEWS CENTER	04/28/92

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*EPP-208 (eff. 07/20/89)	7	ACCESS CONTROL DURING EMERGENCIES	06/10/91
*EPP-209 (eff. 04/11/89)	3	PERSONNEL ASSEMBLY AND ACCOUNTABILITY	Proc.Retired (replaced by EPP-314)
*EPP-210 (eff. 04/11/89)	7	EVACUATION	Proc.Retired (replaced by EPP-314)
*EPP-211	1	RELEASE OF TOXIC MATERIAL	Proc.Retired
*EPP-212	1	RELEASE OF INFORMATION TO THE PUBLIC	Proc.Retired
*EPP-214 (eff. 09/26/90)	1	ACTIVATION AND OPERATION OF THE LOGISTICAL SUPPORT CENTER (LSC)	03/31/92
*EPP-300 (eff. 10/15/87)	3	MANUAL CALCULATION OF OFFSITE DOSE RATES	10/15/91
*EPP-301	2	COMPUTER ASSESSMENT OF RADIOLOGICAL CONDITIONS	Proc.Retired
*EPP-302	4	MANUAL ASSESSMENT OF RADIOLOGICAL CONDITIONS	Proc.Retired
*EPP-303 (eff. 11/09/90)	6	MICRO-COMPUTER BASED, EMERGENCY DOSE ASSESSMENT	09/23/91
*EPP-304 (eff. 07/20/89)	10	PROTECTIVE ACTION RECOMMENDATIONS	11/11/91
*EPP-305 (eff. 07/27/90)	8	EMERGENCY EXPOSURE GUIDELINES AND PERSONNEL DOSIMETRY	02/18/92
*EPP-306 (eff. 03/30/90)	8	USE OF THYROID BLOCKING AGENTS	08/26/91
*EPP-307 (eff. 04/11/89)	5	RADIOLOGICAL MONITORING OF SITE EVACUEES	Proc.Retired (replaced by EPP-314)

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*EPP-309 (eff. 03/23/90)	9	ONSITE/IN-PLANT RADIOLOGICAL SURVEYS AND OFFSITE RADIOLOGICAL MONITORING	09/16/91
*EPP-310 (eff. 03/23/90)	6	ONSITE AND IN-PLANT RADIOLOGICAL SURVEYS	Proc.Retired (replaced by EPP-309 R-9)
*EPP-311 (eff. 01/11/91)	3	ASSESSMENT OF PLANT CONDITIONS DURING DECLARED EMERGENCIES USING THE EMERGENCY RESPONSE FACILITY COMPUTER SYSTEM (ERFCS)	01/06/92
*EPP-312 (eff. 12/01/89)	1	CORE DAMAGE ASSESSMENT	08/23/91
*EPP-313 (eff. 08/10/90)	1	MEDICAL RESPONSE	04/07/92
*EPP-314 (eff. 10/12/90)	2	EVACUATION AND ACCOUNTABILITY	02/28/92
*EPP-900 (eff. 01/11/91)	2	ADMINISTRATIVE CONTROL OF EMERGENCY PLAN, AND ASSOCIATED PROCEDURES, FORMS AND WORKBOOKS	05/05/91
*EPP-901 (eff. 12/15/89)	1	MAINTENANCE AND INVENTORY OF EMERGENCY EQUIPMENT AND SUPPLIES	04/05/92
*EPP-902 (eff. 04/06/90)	2	EMERGENCY PREPAREDNESS TRAINING/DRILLS AND EXERCISES	01/30/92
*EPP-903 (eff. 10/03/89)	0	PUBLIC INFORMATION ACTIVITIES TO SUPPORT EMERGENCY PREPAREDNESS	10/03/91
*EPP-904 (eff. 07/25/90)	1	MAINTENANCE OF THE EMERGENCY PREPAREDNESS PROGRAM	07/20/91
*EPP-905 (eff. 07/20/89)	0	DESIGNATED EMERGENCY USE VEHICLES	07/20/91

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*EPP-906 (eff. 11/26/90)	1	MANAGEMENT OF EMERGENCY PREPAREDNESS COMPUTER SYSTEMS	11/17/91
*EPP-907 (eff. 03/16/90)	0	ALERT AND NOTIFICATION SYSTEM SURVEILLANCE AND REPORTS	03/16/92

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*EPP-102	4	REVIEW, UPDATE AND CONTROL OF THE EMERGENCY PLAN	Proc.Retired
*EPP-103	1	DUTIES OF THE EMERGENCY COORDINATOR	Proc.Retired
*EPP-104 (eff. 01/30/89)	5	EMERGENCY PREPAREDNESS DRILLS AND EXERCISES	Proc.Retired (replaced by EPP-902 R-0)
*EPP-105	1	NOTIFICATION OF OFFSITE OFFICIALS PENDING FLOOD OR FAILURE OF THE SQUAW CREEK RESERVOIR DAM	Proc.Retired
*EPP-106	1	SURVEILLANCE OF EMERGENCY EQUIPMENT	Proc.Retired
*EPP-107 (eff. 04/05/89)	7	MAINTENANCE AND INVENTORY OF EMERGENCY EQUIPMENT AND SUPPLIES	Proc.Retired (replaced by EPP-901 R-0)
*EPP-108 (eff. 07/20/89)	6	CONTROL OF DESIGNATED EMERGENCY USE VEHICLES USED FOR ROUTINE OPERATIONS	Proc.Retired (replaced by EPP-905 R-0)
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*EPP-109 (eff. 05/11/90)	8	DUTIES AND RESPONSIBILITIES OF THE EMERGENCY COORDINATOR/RECOVERY MANAGER	09/18/91
*EPP-110	1	CORPORATE SUPPORT	Proc.Retired
*EPP-111	1	OFFSITE SUPPORT AND ASSISTANCE	Proc.Retired
*EPP-112 (eff. 05/11/90)	7	DUTIES OF CONTROL ROOM PERSONNEL DURING EMERGENCIES	09/16/91
*EPP-113	1	CORPORATE SUPPORT	Proc.Retired
*EPP-114	1	OFFSITE SUPPORT AND ASSISTANCE	Proc.Retired

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Procedure No.	Revision No.	Procedure Title	Review Date
*EPP-115 (eff. 05/05/89)	5	RELEASE OF EMERGENCY-RELATED INFORMATION TO THE PUBLIC	Proc.Retired (replaced by EPP-207 R-5)
*EPP-116 (eff. 05/11/90)	4	EMERGENCY REPAIR AND DAMAGE CONTROL AND IMMEDIATE ENTRIES	09/24/91
*EPP-117 (eff. 03/30/90)	5	ADMINISTRATIVE CONTROLS FOR SAMPLING UNDER POST ACCIDENT CONDITIONS	02/04/92
*EPP-118 (eff. 03/16/90)	4	ALERT AND NOTIFICATION SYSTEM SURVEILLANCE & REPORTS	Proc Retired (replaced by EPP-907 R-0)
*EPP-119 (eff. 03/31/89)	1	SOUND POWERED TELEPHONE SYSTEM AND EMERGENCY EVACUATION ALARM SYSTEM PERIODIC TESTING	Proc.Retired
*EPP-121 (eff. 05/11/90)	6	REENTRY, RECOVERY AND CLOSEOUT	09/14/91
*EPP-201 (eff. 12/22/89)	7	ASSESSMENT OF EMERGENCY ACTION LEVELS, EMERGENCY CLASSIFICATION AND PLAN ACTIVATION	09/17/91
*EPP-202 (eff. 01/11/91)	4	EMERGENCY COMMUNICATIONS SYSTEMS AND EQUIPMENT	08/26/91
*EPP-203 (eff. 05/26/89)	9	NOTIFICATIONS	10/10/91
*EPP-204 (eff. 11/09/90)	8	ACTIVATION AND OPERATION OF THE TECHNICAL SUPPORT CENTER (TSC)	08/08/91
*EPP-205 (eff. 09/28/90)	7	ACTIVATION AND OPERATION OF THE OPERATIONS SUPPORT CENTER (OSC)	09/14/91
*EPP-206 (eff. 11/09/90)	8	ACTIVATION AND OPERATION OF THE EMERGENCY OPERATIONS FACILITY (EOF)	09/16/91
*EPP-207 (eff. 07/20/89)	6	ACTIVATION AND OPERATION OF THE NEWS CENTER	04/28/92

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Procedure No.	Revision No.	Procedure Title	Review Date
*EPP-208 (eff. 07/20/89)	7	ACCESS CONTROL DURING EMERGENCIES	06/10/91
*EPP-209 (eff. 04/11/89)	3	PERSONNEL ASSEMBLY AND ACCOUNTABILITY	Proc.Retired (replaced by EPP-314)
*EPP-210 (eff. 04/11/89)	7	EVACUATION	Proc Retired (replaced by EPP-314)
*EPP-211	1	RELEASE OF TOXIC MATERIAL	Proc Retired
*EPP-212	1	RELEASE OF INFORMATION TO THE PUBLIC	Proc.Retired
*EPP-214 (eff. 09/26/90)	1	ACTIVATION AND OPERATION OF THE LOGISTICAL SUPPORT CENTER (LSC)	03/31/92
*EPP-300 (eff. 10/15/87)	3	MANUAL CALCULATION OF OFFSITE DOSE RATES	10/15/91
*EPP-301	2	COMPUTER ASSESSMENT OF RADIOLOGICAL CONDITIONS	Proc.Retired
*EPP-302	4	MANUAL ASSESSMENT OF RADIOLOGICAL CONDITIONS	Proc.Retired
*EPP-303 (eff. 11/09/90)	6	MICRO-COMPUTER BASED, EMERGENCY DOSE ASSESSMENT	09/23/91
*EPP-304 (eff. 03/15/91)	11	PROTECTIVE ACTION RECOMMENDATIONS	11/11/91
*EPP-305 (eff. 07/27/90)	8	EMERGENCY EXPOSURE GUIDELINES AND PERSONNEL DOSIMETRY	02/18/92
*EPP-306 (eff 03/30/90)	8	USE OF THYROID BLOCKING AGENTS	08/26/91
*EPP-307 (eff. 04/11/89)	5	RADIOLOGICAL MONITORING OF SITE EVACUEES	Proc.Retired (replaced by EPP-314)

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Procedure No.	Revision No.	Procedure Title	Review Date
*EPP-308 (eff. 04/07/89)	6	TRANSPORTING CONTAMINATED INJURED PERSONNEL	Proc Retired (replaced by EPP-313 R-0)
*EPP-309 (eff. 03/23/90)	9	ONSITE/IN-PLANT RADIOLOGICAL SURVEYS AND OFFSITE RADIOLOGICAL MONITORING	09/16/91
*EPP-310 (eff. 03/23/90)	6	ONSITE AND IN-PLANT RADIOLOGICAL SURVEYS	Proc Retired (replaced by EPP-309 R-9)
*EPP-311 (eff. 01/11/91)	3	ASSESSMENT OF PLANT CONDITIONS DURING DECLARED EMERGENCIES USING THE EMERGENCY RESPONSE FACILITY COMPUTER SYSTEM (ERFCS)	01/06/92
*EPP-312 (eff. 12/01/89)	1	CORE DAMAGE ASSESSMENT	08/23/91
*EPP-313 (eff. 08/10/90)	1	MEDICAL RESPONSE	04/07/92
*EPP-314 (eff. 10/12/90)	2	EVACUATION AND ACCOUNTABILITY	02/28/92
*EPP-900 (eff. 01/11/91)	2	ADMINISTRATIVE CONTROL OF EMERGENCY PLAN, AND ASSOCIATED PROCEDURES, FORMS AND WORKBOOKS	05/05/92
*EPP-901 (eff. 12/15/89)	1	MAINTENANCE AND INVENTORY OF EMERGENCY EQUIPMENT AND SUPPLIES	04/05/92
*EPP-902 (eff. 04/06/90)	2	EMERGENCY PREPAREDNESS TRAINING/DRILLS AND EXERCISES	01/30/92
*EPP-903 (eff. 10/03/89)	0	PUBLIC INFORMATION ACTIVITIES TO SUPPORT EMERGENCY PREPAREDNESS	10/03/91
*EPP-904 (eff. 07/25/90)	1	MAINTENANCE OF THE EMERGENCY PREPAREDNESS PROGRAM	07/20/91
*EPP-905 (eff. 07/20/89)	0	DESIGNATED EMERGENCY USE VEHICLES	07/20/91

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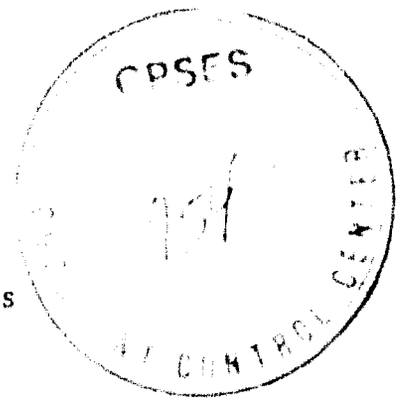
Procedure No.	Revision No.	Procedure Title	Review Date
*EPP-906 (eff. 11/26/90)	1	MANAGEMENT OF EMERGENCY PREPAREDNESS COMPUTER SYSTEMS	11/17/91
*EPP-907 (eff. 03/16/90)	0	ALERT AND NOTIFICATION SYSTEM SURVEILLANCE AND REPORTS	03/16/92

COMANCHE PEAK STEAM ELECTRIC STATION

EMERGENCY PLAN MANUAL

QUALITY-RELATED



PROTECTIVE ACTION RECOMMENDATIONS

PROCEDURE NO EPP-304

REVISION NO 11

SORC MEETING NO [illegible] DATE [illegible]

EFFECTIVE DATE [illegible]

APPROVED BY [signature] for AAS DATE 3/11/91

VICE PRESIDENT, NUCLEAR OPERATIONS

1.0 Purpose

This procedure describes when and how protective action recommendations are made to appropriate local and State agencies during a declared emergency at CPSES.

2.0 Applicability

This procedure applies to the Emergency Coordinator, the Shift Technical Advisor, On-shift Radiation Protection Technician, Radiation Protection Coordinator, Onsite and Offsite Radiological Assessment Coordinators and other personnel in the Emergency Organization designated by the Emergency Coordinator to assist in preparing and issuing protective action recommendations.

3.0 Definitions

3.1 Baseline Recommendation - A protective action recommendation based upon a given emergency classification. Recommended when time is not available to evaluate radioactive release or core/containment conditions.

3.2 Emergency Planning Zone (EPZ) - That area surrounding the Station in which emergency preparedness planning is conducted for the protection of the public.

3.1.1 10 Mile - That area, approximately 10 miles in radius from the center of CPSES, for which emergency planning considerations of the plume exposure pathway have been given in order to assure that prompt and effective actions can be taken to protect the public and property in the event of an accident.

3.2.2 50 Mile - That area, approximately 50 miles in radius from the center of CPSES, for which emergency planning considerations of the ingestion exposure pathway have been given in order to assure that effective actions can be taken to protect the public.

3.3 Emergency Response Zones (ERZ) - Predesignated areas within the 10 mile EPZ used to identify affected areas for protective actions. These 27 alpha-numerically designated zones use existing roads and natural formations (i.e., rivers and lakes) to form their boundaries. Besides these alpha-numerical zones, three additional zones: Glen Rose, Granbury and Tolar, have been established and use the town limits as their boundaries. The Emergency Response Zones are shown in Attachment 5, "Emergency Response Zone Map."

3.4 Minimum Affected Area - The minimum affected area is defined as that sector containing the centerline of the plume and the contiguous 22.5° sectors on either side of the plume centerline sector.

3.5 Projected Dose (Exposure Projection) - A calculated or expected whole body or child thyroid exposure that would be received by individual(s) if no protective actions are taken.

3.6 Protective Actions - Those emergency measures taken to prevent or minimize radiological exposures to affected population groups.

3.7 Protective Action Guides (PAGs) - Projected radiological doses to individuals in the general population that warrant protective action to reduce or avoid those doses

3.8 Protective Action Recommendation (PAR) - Recommended protective measures for the plume exposure pathway, such as sheltering and/or evacuating from an Emergency Response Zone (ERZ), given to offsite agencies relative to protective action guide values, an emergency class or plant conditions.

4.0 Instructions

4.1 Responsibilities

4.1.1 When an emergency classification is made and the Shift Supervisor assumes the role of Emergency Coordinator, he/she shall be responsible with the assistance of the Shift Technical Advisor to review offsite dose calculations performed by the On-Shift Radiation Protection Technician and develop protective action recommendations.

NOTE: The responsibility to approve and issue protective action recommendations shall remain with the Emergency Coordinator regardless of his/her physical location.

4.1.2 The On-Shift Radiation Protection Technician should continually assess potential and actual radionuclide releases, conduct and review dose calculations and keep the Shift Supervisor informed of his/her activities

4.1.3 When the role of Emergency Coordinator is transferred to the TSC Manager, he/she will be dependant upon the Onsite Radiological Assessment Coordinator to provide offsite dose calculations and protective action recommendations for his/her review and consideration. The actual calculations for these will be performed by the TSC RMS Computer Operators. Should EOF personnel become able to perform their functions prior to activation of the EOF, the TSC Manager as Emergency Coordinator may transfer support

functions to the EOF, including dose assessment and development of protective action recommendations.

4.1.4 When the role of Emergency Coordinator is transferred from the TSC Manager to the EOF Manager, he/she will be dependant upon the Offsite Radiological Assessment Coordinator to provide offsite dose calculations and protective action recommendations through the Radiation Protection Coordinator for his/her review and consideration. The actual calculations for these will be performed by the EOF RMS Computer Operators.

4.1.5 If time permits, the Texas Department of Health, Bureau of Radiation Control (BRC), should be offered the opportunity to review protective action recommendations prior to their transmittal to offsite agencies. Except, Control Room communicators shall not contact nor be contacted by the BRC for this review. In the Technical Support Center (TSC), the review is offered via the telephone to the BRC's designated Chief, Field Operations or designee by the TSC HP Network Communicator under the direction of the TSC Onsite Radiological Assessment Coordinator (ONRAC) In the Emergency Operations Facility (EOF), as soon as PARs are available, they should be discussed between the OFFRAC and BRC's Chief of Field Operations or his designee located in Classroom 6. Once the PARs are written on the notification message form, a final review is offered by the EOF Communications Coordinator under the direction of the EOF Radiation Protection Coordinator to the BRC's Chief of Field Operations.

4.2 Precautions

4.2.1 The Emergency Coordinator shall provide protective action recommendations to county and State agencies as quickly as possible but in no case later that 15 minutes after classifying or reclassifying an emergency.

4.2.2 County officials have final decision-making authority for implementation of protective actions. State officials, upon request from county officials, provide assistance as needed. TU Electric has no authority to impose protective actions beyond its site boundary.

4.2.3 The minimum affected area should be expanded to include two (2) additional 22.5° sectors one on either side of original set (for a total of five (5) sectors) if any of the following conditions exits:

1. Onsite wind direction is not available;
2. Pasquill Stability Class of A or B exists;

Additional sectors and corresponding Emergency Response Zones within these sectors should be determined using the map shown on Attachment 5, "Emergency Response Zone Map."

4.3 Initial Actions

4.3.1 Obtain the emergency classification from the Emergency Coordinator.

4.3.2 Refer to Attachment 1, "Protective Action Recommendation (PAR) Flow Chart," to determine the appropriate PAR. This includes making a baseline recommendation.

4.3.3 If time permits and the facility Emergency Coordinator duties are not in the Control Room, contact the Texas Department of Health, Bureau of Radiation Control (BRC) and provide a brief explanation of the emergency situation, PARs being recommended and the basis for the PARs.

4.3.4 Provide PARs to the Emergency Coordinator.

4.4 Subsequent Actions

4.4.1 Field survey findings acquired in accordance with EPP-309, "Onsite/In-plant Radiological Surveys and Offsite Radiological Monitoring" should be used to verify or update previously determined protective action recommendations.

4.4.1.1 If the micro-computer method of dose assessment is available then enter the data into the Emergency Dose Assessment Modal (EDAM) program to obtain whole body doses and child thyroid doses and compare this information with total doses obtained using onsite release rate information.

4.4.1.2 If the micro-computer is not available and the manual method of dose assessment is used then take the field survey radioiodine air concentration and estimate the child thyroid dose relative to a given release duration using Attachment 6, "Projected Child Thyroid Dose Chart." Also, multiply the field dose rate by the release duration to obtain the total whole body dose. Total thyroid and whole body dose obtained in this manner may be compared with manual dose assessment total doses for comparable distances from the plant.

4.4.3 Protective Action Recommendations (PARs) should be reevaluated whenever any of the following conditions occur:

4.4.3.1 When a followup notification message is being sent to State and county agencies.

4.4.3.2 Whenever the emergency classification is escalated

4.4.3.3 Whenever dose projections are updated in accordance with applicable procedures.

4.4.3.4 Whenever plant status changes would affect the health and safety of the public.

5.0 References

5.1 CPSES Emergency Plan

5.2 State of Texas Emergency Management Plan

5.3 Hood County Emergency Operations Plan

5.4 Somervell County Emergency Operations Plan

5.5 EPA-520/1-75-001, "Manual of Protective Action Guides and Protective Action for Nuclear Incidents", U.S. Environmental Protective Agency, September, 1975 (revised June, 1980).

5.6 EPP-201, "Assessment of Emergency Action Levels, Emergency Classifications and Plan Activation"

5.7 EPP-203, "Notifications"

5.8 EPP-300, "Manual Calculation of Offsite Dose Rates"

5.9 EPP-303, "Micro-Computer Based, Emergency Dose Assessment"

5.10 EPP-306, "Use of Thyroid Blocking Agents"

5.11 EPP-309, "Onsite/In-plant Radiological Surveys and Offsite Radiological Monitoring"

5.12 IE Information Notice No. 83-28, "Criteria for Protective Action Recommendations for General Emergencies"

5.13 TNO 89-0413, "Assumptions Used in the Development of the Protective Action Recommendations Flow Chart"

6.0 Attachments

6.1 Attachment 1, "Protective Action Recommendation Flowchart"

6.2 Attachment 2, " Radiological Release in Progress Flowchart"

6.3 Attachment 3, "Core/Containment Conditions Flowchart"

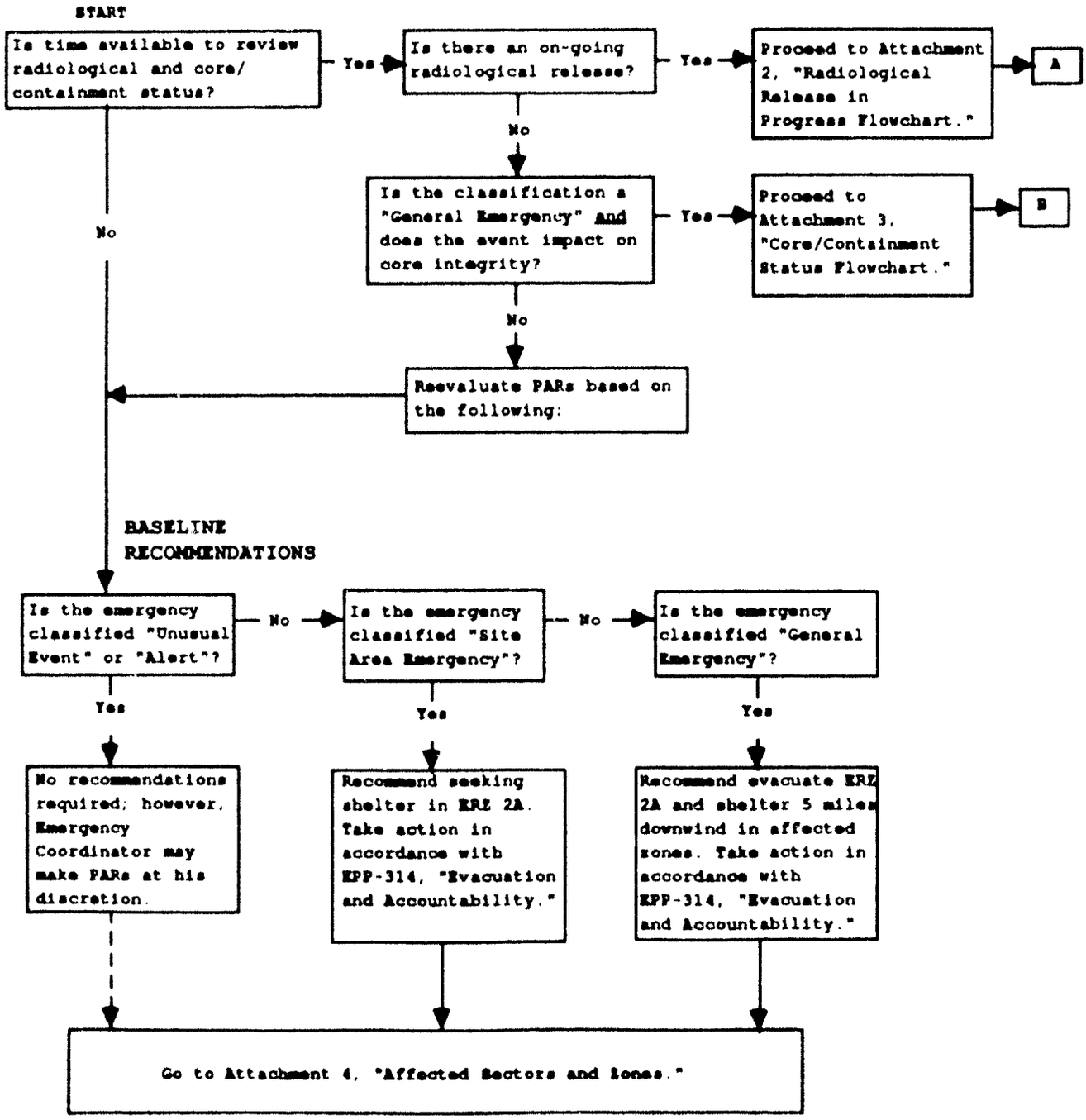
6.4 Attachment 4, "Affected Sectors and Zones"

6.5 Attachment 5, "Emergency Response Zone Map"

6.6 Attachment 6, "Projected Child Thyroid Dose Chart"

ATTACHMENT 1
PAGE 1 OF 1

PROTECTIVE ACTION RECOMMENDATION FLOWCHART
(PLUME EXPOSURE PATHWAY)



ATTACHMENT 2
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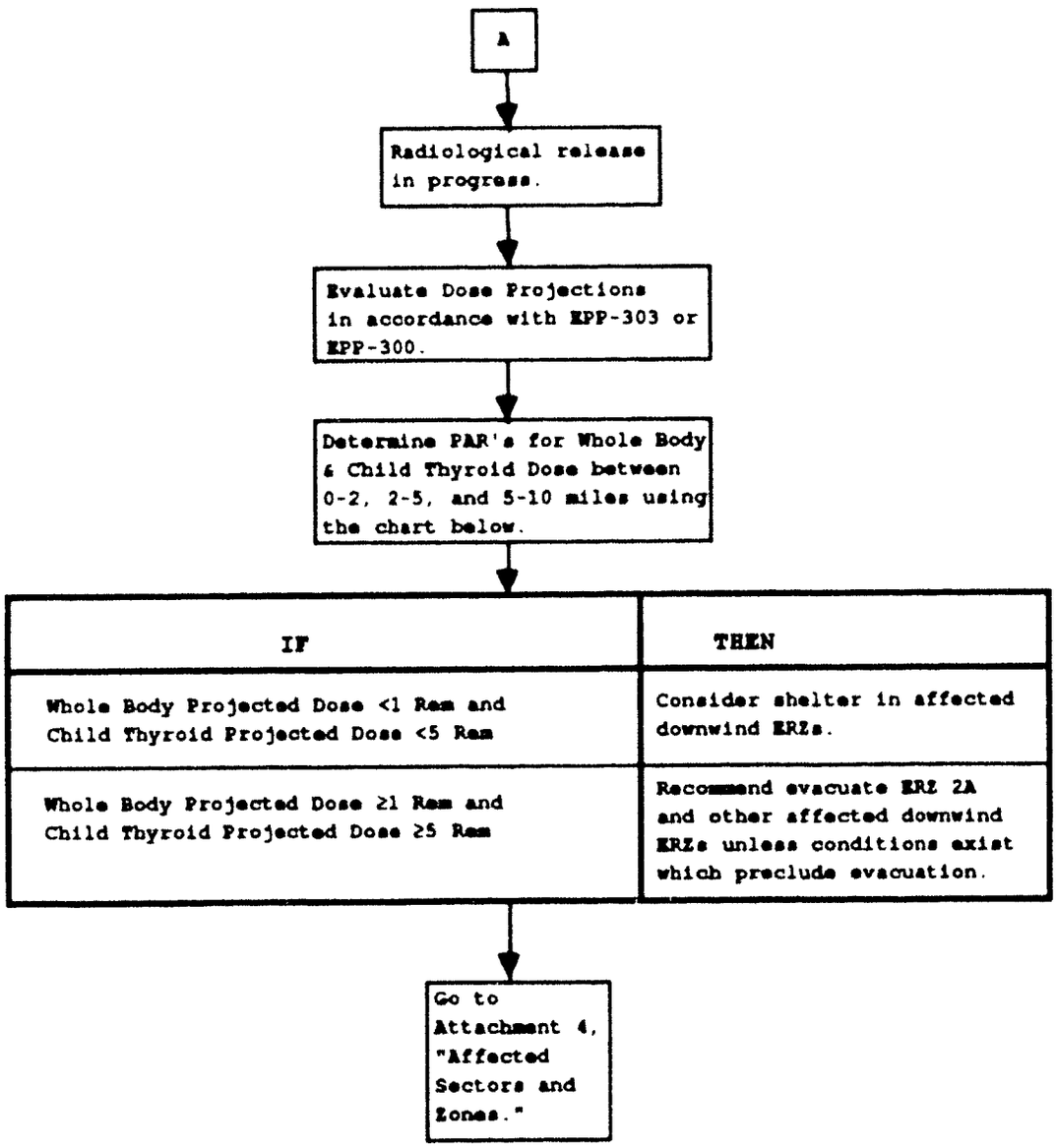
RADIOLOGICAL RELEASE IN PROGRESS FLOWCHART



ATTACHMENT 3
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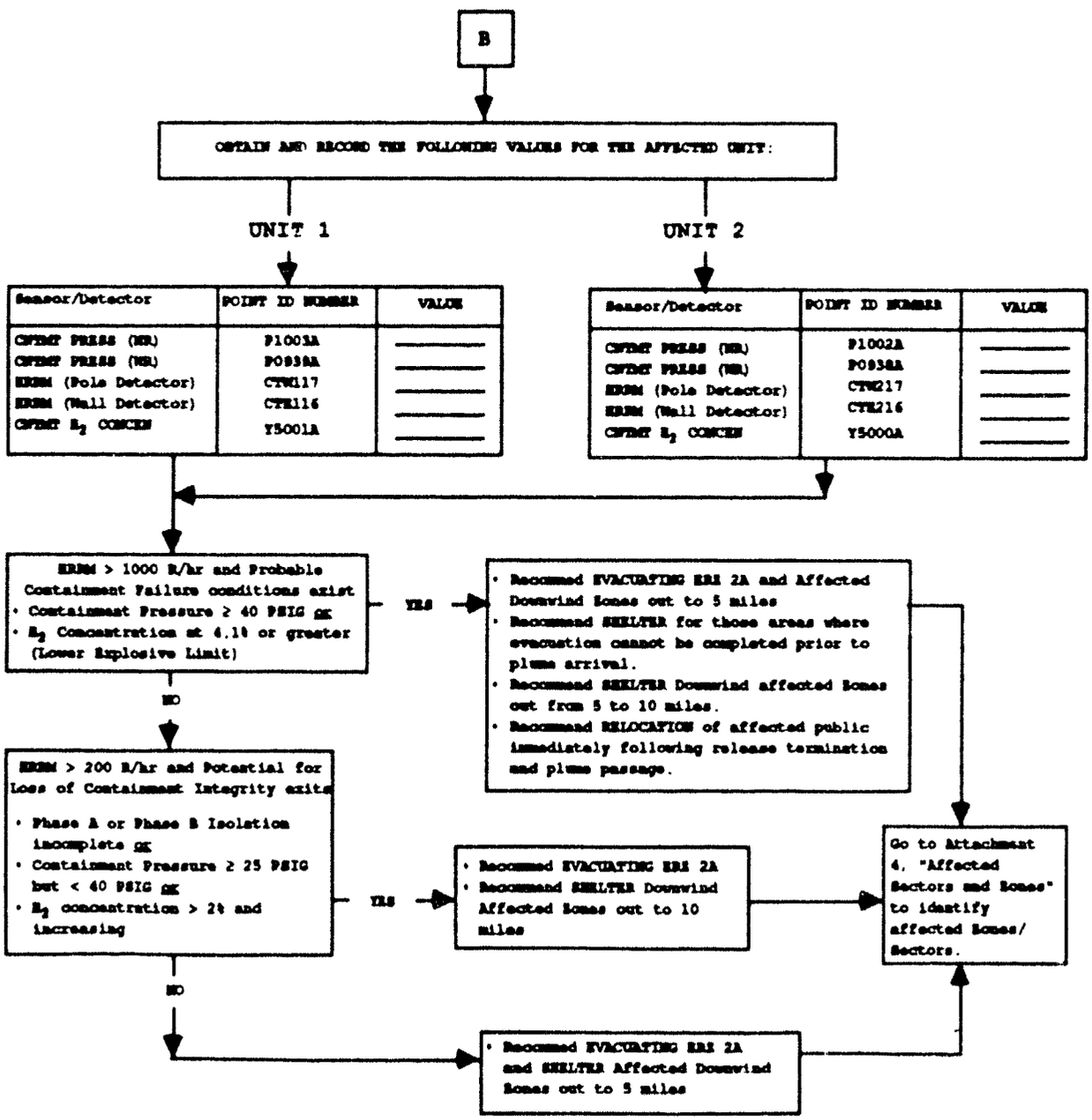
CORE/CONTAINMENT CONDITIONS FLOWCHART



ATTACHMENT 4
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AFFECTED SECTORS AND ZONES

This Job Aid permits the user to identify Emergency Response Zones (ERZs) IF (1) wind direction is known and (2) downwind distance from the plant is known.

STEP 1: Enter wind direction: _____° (degrees)

CAUTION: If onsite wind direction is not available or Pasquill Stability Class A or B exists the minimum affected area should be expanded to include two (2) additional 22.5° sectors one on either side of original set (for a total of five (5) sectors). Add any additional ERZ's from the two adjoining sectors to your new listing of affected ERZ's.

Met Source: Primary Met Tower/Backup Met Tower/National Weather Service (circle one)

Pasquill Stability Class _____

STEP 2: Identify Centerline Section Affected using the table below:

IF WIND DIRECTION (degrees) is (From)	THEN CENTERLINE Sector is	IF WIND DIRECTION (degrees) is (From)	THEN CENTERLINE Sector is
348.75 - 11.25	J	168.75 - 191.25	A
11.25 - 33.75	K	191.25 - 213.75	B
33.75 - 56.25	L	213.75 - 236.25	C
56.25 - 78.75	M	236.25 - 258.75	D
78.75 - 101.25	N	258.75 - 281.25	E
101.25 - 123.75	P	281.25 - 303.75	F
123.75 - 146.25	Q	303.75 - 326.25	G
146.25 - 168.75	R	326.25 - 348.75	H

STEP 3: Knowing Centerline Sector, identify affected sectors and Emergency Response Zones (ERZs) using the job aid on this Attachment pages 2, 3 and 4.

STEP 4: Go to Section 4.3.3 in the body of the procedure.

ATTACHMENT 4
PAGE 2 OF 4

AFFECTED SECTORS AND ZONES

IF Centerline Sector is	THEN Affected Sectors are	AND Distance from the Plant is	THEN Emergency Response Zones are
A	RAB	0-2 mi	2A
		>2 mi <5 mi	4B, 4C, 4A, 1B
		>5 mi <10 mi	Granbury, 1D, 4E, 4F, 1C
B	ABC	0-2 mi	2A
		>2 mi <5 mi	4A, 4B, 1B, 1A
		>5 mi <10 mi	Granbury, 1C, 1D, 4E, 2G
C	BCD	0-2 mi	2A
		>2 mi <5 mi	4A, 1A, 1B, 4B, 2B, 2D, 2E
		>5 mi <10 mi	1C, 2G, 1D
D	CDE	0-2 mi	2A
		>2 mi <5 mi	2B, 1A, 2D, 2E, 4A, 1B
		>5 mi <10 mi	1C, 2G, 2E, 2F
E	DEF	0-2 mi	2A
		>2 mi <5 mi	2B, 2D, 2E, 1A
		>5 mi <10 mi	2E, 2F, 1C, 2G, 2H

ATTACHMENT 4
PAGE 3 OF 4

AFFECTED SECTORS AND ZONES

IF Centerline Sector is	THEN Affected Sectors are	AND Distance from the Plant is	THEN Emergency Response Zones are
F	EFG	0-2 mi	2A
		>2 mi <5 mi	Glen Rose, 2B, 2D, 2E, 2C
		>5 mi <10 mi	2H, 2F, 2J
G	FGH	0-2 mi	2A
		>2 mi <5 mi	Glen Rose, 2C, 2B, 2D
		>5 mi <10 mi	2H, 2J, 3C
H	GHJ	0-2 mi	2A
		>2 mi <5 mi	Glen Rose, 2C, 3B
		>5 mi <10 mi	2J, 3C, 2H, 3D, 3F
J	HJK	0-2 mi	2A
		>2 mi <5 mi	Glen Rose, 2C, 3B
		>5 mi <10 mi	3C, 3D, 3F, 2J
K	JKL	0-2 mi	2A
		>2 mi <5 mi	Glen Rose, 3B, 2C, 3A
		>5 mi <10 mi	3D, 3F, 3C, 3E

ATTACHMENT 4
PAGE 4 OF 4

AFFECTED SECTORS AND ZONES

IF Centerline Sector is	THEN Affected Sectors are	AND Distance from the Plant is	THEN Emergency Response Zones are
L	KLM	0-2 mi	2A
		>2 mi <5 mi	3A, 3B, 4D
		>5 mi <10 mi	3D, 3E, 3F, 4H
M	LMN	0-2 mi	2A
		>2 mi <5 mi	3A, 4D, 4C, 3B
		>5 mi <10 mi	3E, 4H, 3D, 4G
N	MNP	0-2 mi	2A
		>2 mi <5 mi	4C, 4D, 3A
		>5 mi <10 mi	4G, 4H, 3E, 4F
P	NPQ	0-2 mi	2A
		>2 mi <5 mi	4C, 4D
		>5 mi <10 mi	Tolar, 4F, 4G, 4H
Q	PQR	0-2 mi	2A
		>2 mi <5 mi	4C, 4B
		>5 mi <10 mi	Tolar, 4F, 4G, 4E
R	QRA	0-2 mi	2A
		>2 mi <5 mi	4B, 4C
		>5 mi <10 mi	Granbury, Tolar, 4E, 4F, 1D

ATTACHMENT 5
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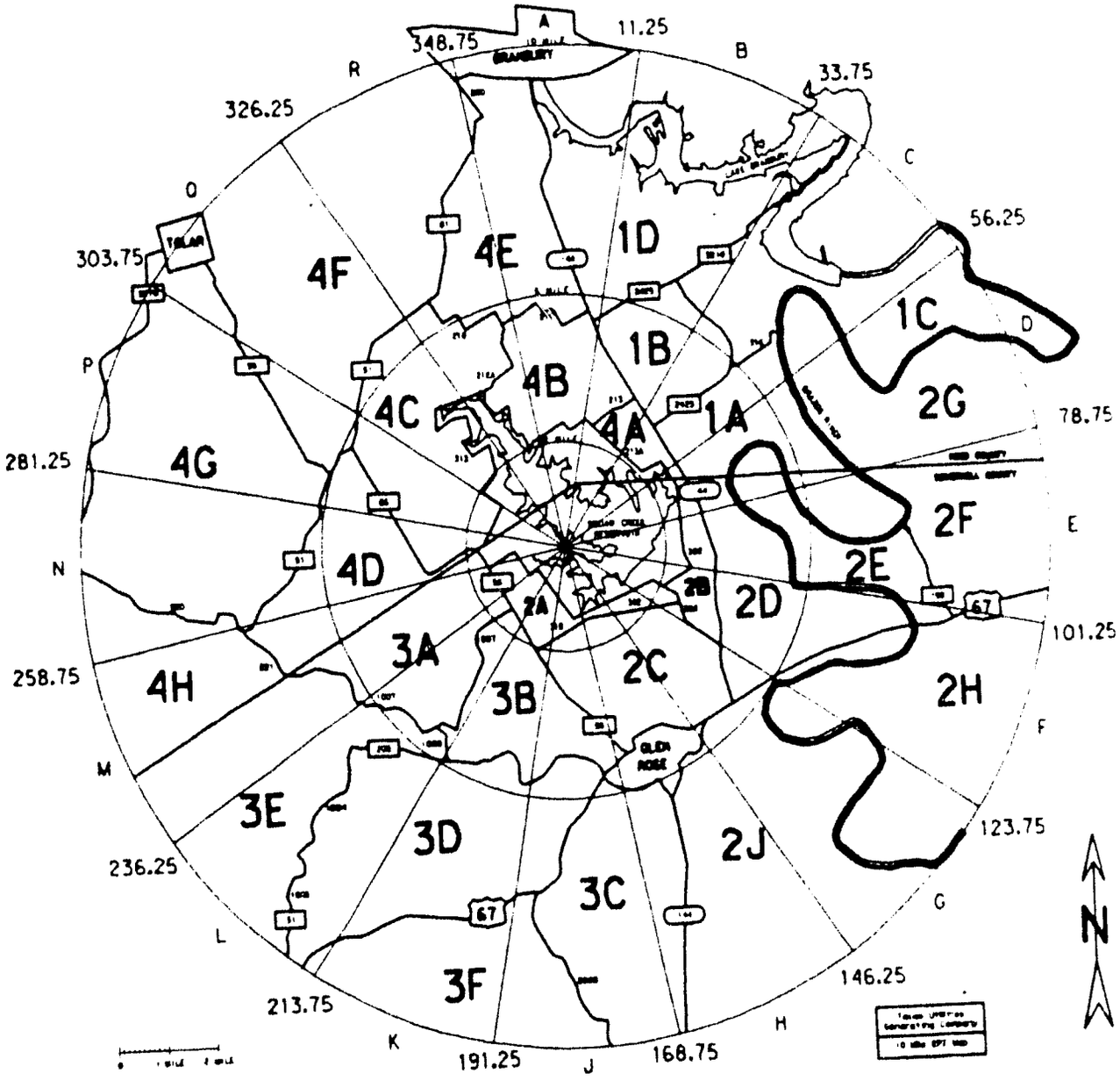
EMERGENCY RESPONSE ZONE MAP



ATTACHMENT 6
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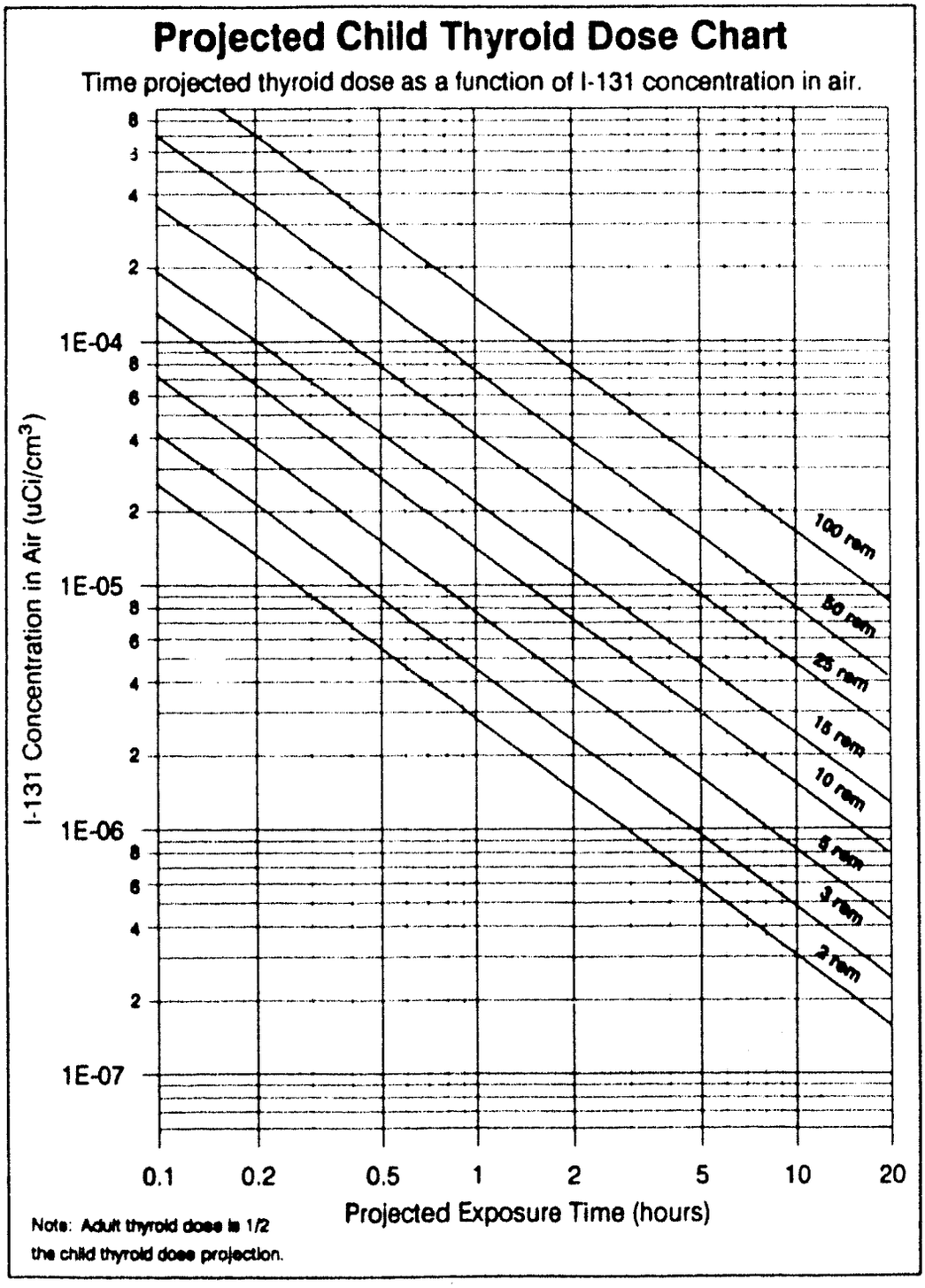
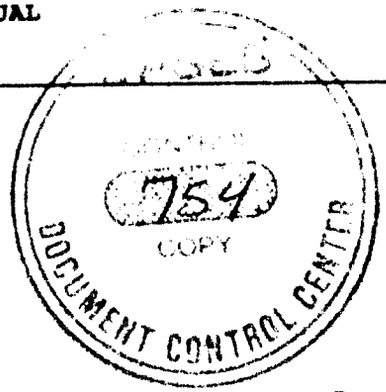
PROJECTED CHILD THYROID DOSE CHART



COMANCHE PEAK STEAM ELECTRIC STATION

EMERGENCY PLAN FORM MANUAL



FORM INDEX

REVISION 28

Form No.	Revision No.	Form Title	Review Date
*EPP-109-1 (eff. 05/11/90)	4	EMERGENCY ORGANIZATION ACTIVITIES LOG SHEET	05/11/92
*EPP-109-2 (eff. 05/11/90)	5	EMERGENCY COORDINATOR CHECKLIST (NOTIFICATION OF UNUSUAL EVENT)	05/11/92
*EPP-109-3 (eff. 05/11/90)	5	EMERGENCY COORDINATOR CHECKLIST (ALERT)	05/11/92
*EPP-109-4 (eff. 05/11/90)	5	EMERGENCY COORDINATOR CHECKLIST (SITE AREA EMERGENCY OR GENERAL EMERGENCY)	05/11/92
*EPP-116-1 (eff. 05/11/90)	4	EMERGENCY WORK PERMIT	05/26/92
*EPP-116-2 (eff. 05/11/90)	0	EMERGENCY WORK PERMIT ISSUE LOG	05/11/92
*EPP-117-1 (eff. 03/30/90)	2	REQUEST FOR POST ACCIDENT SAMPLE	03/30/92
*EPP-121-1 (eff. 05/11/90)	4	RECOVERY MANAGER CHECKLIST	05/11/92
*EPP-203-8 (eff. 05/26/89)	1	NOTIFICATION MESSAGE FORM	05/26/92
*EPP-203-9 (eff. 07/24/89)	0	NRC UPDATES	07/24/91
*EPP-204-1 (eff. 11/09/90)	6	TECHNICAL SUPPORT CENTER CHECKLIST	11/09/91
*EPP-204-4 (eff. 11/09/90)	0	COMANCHE PEAK STEAM ELECTRIC STATION OPERATIONAL TREND SHEET	11/09/91
*EPP-204-5 (eff. 11/09/90)	0	COMANCHE PEAK STEAM ELECTRIC STATION RADIOLOGICAL TREND SHEET	11/09/91
*EPP-205-2 (eff. 09/28/90)	1	TEAM ASSIGNMENT WORKSHEET	09/28/91

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REVISION 28

Form No.	Revision No.	Form Title	Review Date
*EPP-205-3 (eff. 09/28/90)	0	AUXILIARY OPERATORS' WORK ACTIVITY LOG SHEET	09/28/91
*EPP-206-1 (eff. 11/09/90)	6	EMERGENCY OPERATIONS FACITLITY (EOF) CHECKLIST	05/26/92
*EPP-206-2 (eff. 07/20/89)	0	RADIATION PROTECTION SIGN-OFF SHEET	07/20/91
*EPP-207-1 (eff. 07/20/89)	1	NEWS CENTER ACTIVATION CHECKLIST	07/20/91
*EPP-207-2 (eff. 07/20/89)	1	NEWS MEDIA REGISTRATION LOG	07/20/91
*EPP-207-6 (eff. 07/20/89)	0	ALTERNATE NEWS CENTER AND ANNEX ACTIVATION CHECKLIST	07/20/91
*EPP-207-7 (eff. 07/20/89)	0	RUMOR CONTROL/MEDIA RESPONSE CALL SHEET	07/20/91
*EPP-207-8 (eff. 07/20/89)	0	MEDIA MONITORING INFORMATION	07/20/91
*EPP-207-9 (eff. 07/20/89)	0	NEWS CONFERENCE RECORDING CHECK OUT LIST	07/20/91
*EPP-207-10 (eff. 07/20/89)	0	EMERGENCY-RELATED PUBLIC INFORMATION	07/20/91
*EPP-207-11 (eff. 07/20/89)	0	INSTRUCTION TO THE NEWS MEDIA	07/20/91
*EPP-214-1 (eff. 09/26/90)	1	LOGISTICAL SUPPORT CENTER CHECKLIST	09/26/91
*EPP-214-2 (eff. 09/26/90)	1	EMERGENCY ORGANIZATION MANPOWER ACCOUNTABILITY/SHIFT ASSIGNMENT ROSTER	09/26/91
*EPP-303-1 (eff. 11/09/90)	5	STACK RELEASE CALCULATION	11/09/91

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REVISION 28

Form No.	Revision No.	Form Title	Review Date
*EPP-303-2 (eff. 11/09/90)	5	CONTAINMENT LEAKAGE CALCULATION	11/09/91
*EPP-303-3 (eff. 11/09/90)	5	STEAM GENERATOR TUBE LEAK CALCULATION	11/09/91
*EPP-303-4 (eff. 11/09/90)	0	SURVEY DATA OPTION	11/09/91
*EPP-305-2 (eff. 07/27/90)	6	EMERGENCY EXPOSURE AUTORIZATION	07/27/91
*EPP-306-2 (eff. 03/30/90)	2	RECORD OF POTASSIUM IODIDE DISTRIBUTION	03/30/92
*EPP-309-1 (eff. 03/23/90)	6	RADIOLOGICAL MONITORING DATA SHEET	03/23/92
*EPP-309-2 (eff. 03/23/90)	4	COMMUNICATOR'S RADIOLOGICAL MONITORING DATA SHEET	03/23/92
*EPP-309-3 (eff. 03/23/90)	5	DOSE RATE MEASUREMENT DATA SHEET	03/23/92
*EPP-309-4 (eff. 03/23/90)	5	COMMUNICATOR'S DOSE RATE MEASUREMENT DATA SHEET	03/23/92
*EPP-312-1 (eff. 12/01/89)	1	SAMPLE ANALYSIS	12/01/91
*EPP-312-2 (eff. 12/01/89)	1	CORE INVENTORY	12/01/91
*EPP-312-4 (eff. 12/01/89)	1	DECAY CORRECTION FOR NUCLIDES WITH SIGNIFICANT PERCURSORS	12/01/91
*EPP-312-5 (eff. 12/01/89)	1	CORE ACTIVITY RELEASED	12/01/91
*EPP-312-7 (eff. 12/01/89)	1	AUXILIARY INDICATORS OF CORE DAMAGE	12/01/91

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REVISION 28

Form No.	Revision No.	Form Title	Review Date
*EPP-312-8 (eff. 12/01/89)	0	SUMMARY	12/01/91
*EPP-313-1 (eff. 08/10/90)	1	DOCUMENTING RESPONSE TO AN INJURED PERSON PART I OF III, RECORDING INITIAL INFORMATION	08/10/91
*EPP-314-1 (eff. 10/12/90)	2	EMERGENCY RESPONSE FACILITY (ERF) ACCOUNTABILITY	07/13/91
*EPP-314-2 (eff. 07/13/89)	0	VEHICLE CONTAMINATION RECORD	07/13/91
*EPP-900-3 (eff. 01/11/91)	2	DETERMINATION OF A DECREASE IN THE EFFECTIVENESS OF THE CPSES EMERGENCY PLAN IN ACCORDANCE WITH 10 CFR 50.54(q)	06/14/91
*EPP-900-4 (eff. 01/11/91)	1	EMERGENCY PLANNING WORKBOOK REVISION VERIFICATION	07/18/91
*EPP-901-1 (eff. 12/15/89)	1	INVENTORY SUMMARY	12/15/91
*EPP-901-2 (eff. 12/15/89)	1	VERIFICATION OF OPERABILITY OF DESIGNATED EMERGENCY EQUIPMENT	12/15/91
*EPP-901-3 (eff. 12/15/89)	1	MONTHLY TEST OF SPECIAL TELEPHONE CIRCUITS	12/15/91
*EPP-904-1 (eff. 07/25/90)	1	EMERGENCY PREPAREDNESS TELECOPIER OPERATIONAL CHECKS	07/20/91
*EPP-904-2 (eff. 07/25/90)	0	EMERGENCY PREPAREDNESS TELEPHONE NUMBER VERIFICATION	07/25/91
*EPP-906-1 (eff. 11/17/89)	0	ERROR REPORT FOR EMERGECNCY PREPAREDNESS QUALITY RELATED COMPUTER SYSTEM	11/17/91
*EPP-907-1 (eff. 03/16/90)	0	ALERT AND NOTIFICATION SYSTEM OPERABILITY TEST REPORT	03/16/92

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REVISION 28

Form No.	Revision No.	Form Title	Review Date
*EPP-907-2 (eff. 03/16/90)	0	ALERT AND NOTIFICATION SYSTEM OBSERVERS OPERABILITY TEST REPORT	03/16/92
*EPP-907-3 (eff. 03/16/90)	0	ALERT AND NOTIFICATION SYSTEM MAINTENANCE AND OPERABILITY TRACKING SYSTEM	03/16/92
*EPP-907-4 (eff. 03/16/90)	0	ALERT AND NOTIFICATION SYSTEM OPERABILITY COUNTER AND GROWL TEST REPORT	03/16/92

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON D C 20555

Reply to:
[illegible]
Las Vegas, Nevada [illegible]
Tel: [illegible]
FTS: [illegible]

PLEASE NOTE CHANGE OF ADDRESS

TO: Joseph Holonich, Acting Director, HLPD
Division of High-Level Waste Management, MS 4 H 3

FROM: Nancy J. White, Secretary to
Paul T. Prestholt, Sr. On-Site Licensing Representative

DATE: March 19, 1991

SUBJECT: Paul T. Prestholt YMP Site Report

This report was typed when Mr. Prestholt was away from the office on travel. He has not seen the printout; if there are any changes to be made you will be notified.

njw

1/.

102
WM-11
NH03

9103280160 910319
PDR WASTE PDR
WM-11

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON D C 20555

Reply to:

301 E. Stewart Ave., #203
Las Vegas, Nevada 89101
Tel: (702) 388-6125
FTS: 598-6125

M E M O R A N D U M

DATE: March 19, 1991
FOR: Joseph Holonich, Acting Director, HLPD
Division of High-Level Waste Management, M/S 4 H 3
FROM: Paul T. Prestholt, Sr. OR - YMP
SUBJECT: YMP Site Report for the months of January and February, 1991

1. EXPLORATORY SHAFT FACILITY (ESF) ALTERNATIVES

The following is a key activities schedule for ESF Title II design resumption:

- SNL submitted ESFAS findings report for Project Office acceptance - 12-21-90
- Project Office management review completed 1-5-91
- Revised ESFAS report due Project Office 1-10-91
- Recommendation to RW-1 1-14-91
- RW-1 Decision on ESF Configuration 1-28-91
- Draft Study Report 3-26-91

The above dates (except the last) have been met.

The findings of the ESF alternatives study were (from January TPO meeting):

FINDINGS OF ESF ALTERNATIVES STUDY

- The study considered and screened a large number of alternatives to produce 34 ESF repository options which were then formally evaluated against a wide range of criteria.

- The rank order of the options was determined primarily from the relative probabilities assessed for programmatic viability. Other key measures, such as regulatory approval, likelihood of repository closure, postclosure performance and characterization testing were considered in assessing programmatic viability.

- The top ranked option is consistent with the value judgement expressed by the management panel and the technical judgements expressed by all three members of the technical panels. Only one technical panel member provided a view that produces a different ranking. Even under this view, many of the same options are concluded to be highly rated.

- A number of design features were identified that appear to enhance the overall performance of particular options.

Inserted is the rank order of options under various majority/minority reports. (See Insert A and B).

In accepting the Preliminary Finding Report submitted by the Yucca Mountain Project Office (YMPO), DOE-Hq directed that the Project proceed with the design study focusing on the favorable features of the highest ranked ESF alternatives; proceed with a design study based on post-1988 data, the ESF study and the Calico Hills study, and providing the flexibility to penetrate the Calico Hills unit in the first phase. YMPO is to prepare plans for a phased approach to design development and ESF implementation preserving flexilility to take advantage of findings as data aquisition proceeds. This activity is to be completed by November, 1992.

RANK ORDER OF OPTIONS UNDER VARIOUS MAJORITY/MINORITY REPORTS

MAJORITY BEST JUDGEMENT RANKING		MINORITY VIEW FOR PROGRAM VIABILITY		MINORITY EFN VIEW #1 (7 EXPERTS)		MINORITY EFN VIEW #2 (2 EXPERTS)		MINORITY VIEW ON RETRIEVAL (NO CHANGE)		REV. ESTIMATES FOR TESTING PROBABILITIES	
30	1st	13	1st	30	1st	23	1st	30	1st	30	1st
23	2nd	2	2nd	13	2nd	24	2nd	23	2nd	23	2nd
24	3rd	6	3rd	23	3rd	6	3rd	24	3rd	24	3rd
13	4th	23	4th	24	4th	30	4th	13	4th	13	4th
6	5th	19	5th	7	5th	7	5th	6	5th	6	5th
7	6th	4	6th	6	6th	2	6th	7	6th	7	6th
2	7th	7	7th	4	7th	13	7th	2	7th	2	7th
19	8th	5	8th	19	8th	19	8th	19	8th	19	8th
4	9th	21	9th	2	9th	25	9th	25	9th	4	9th
25	10th	24	10th	25	10th	28	10th	4	10th	25	10th
21	11th	15	11th	21	11th	21	11th	21	11th	21	11th
28	12th	12	12th	28	12th	32	12th	28	12th	28	12th
22	13th	3	13th	22	13th	27	13th	22	13th	22	13th
29	14th	20	14th	29	14th	4	14th	29	14th	29	14th
32	15th	29	15th	8	15th	20	15th	32	15th	32	15th
20	16th	32	16th	32	16th	22	16th	27	16th	27	16th
27	17th	14	17th	20	17th	29	17th	20	17th	20	17th
8	18th	22	18th	27	18th	8	18th	8	18th	8	18th
32	19th	28	19th	33	19th	15	19th	31	19th	31	19th
15	20th	31	20th	5	20th	33	20th	15	20th	33	20th
33	21st	30	21st	15	21st	31	21st	33	21st	15	21st

INSERT A

RANK ORDER OF OPTIONS UNDER VARIOUS MAJORITY/MINORITY REPORTS

(CONTINUED)

MAJORITY BEST JUDGEMENT RANKING	MINORITY VIEW FOR PROGRAM VIABILITY	MINORITY EFN VIEW #1 (7 EXPERTS)	MINORITY EFN VIEW #2 (2 EXPERTS)	MINORITY VIEW ON RETRIEVAL (NO CHANGE)	REV. ESTIMATES FOR TESTING PROBABILITIES
5 22nd	8 22nd	31 22nd	16 22nd	5 22nd	5 22nd
12 23rd	25 23rd	3 23rd	5 23rd	12 23rd	16 23rd
16 25th	11 25th	16 25th	1 25th	3 25th	3 25th
11 26th	33 26th	11 26th	12 26th	11 26th	11 26th
1 27th	18 27th	1 27th	3 27th	1 27th	1 27th
14 28th	1 28th	18 28th	10 28th	14 28th	14 28th
10 29th	17 29th	14 29th	18 29th	10 29th	10 29th
17 30th	10 30th	10 30th	17 30th	17 30th	17 30th
18 31st	27 31st	17 31st	14 31st	18 31st	18 31st
34 32nd	34 32nd	34 32nd	34 32nd	34 32nd	34 32nd
26 33rd	9 33rd	26 33rd	26 33rd	26 33rd	26 33rd
9 34th	26 34th	9 34th	9 34th	9 34th	9 34th

INSERT B

Further, YMPO is to prepare a plan to resolve the NRC's Site Characterization Analysis (SCA) design control objection.

Title II design on alternative 3D is scheduled to start on October 1, 1991. Inserted is a depiction of the Reference Design Concept. Also inserted is a chart showing the integration of alternatives 3D, 24, and 23. (See Inserts C and D).

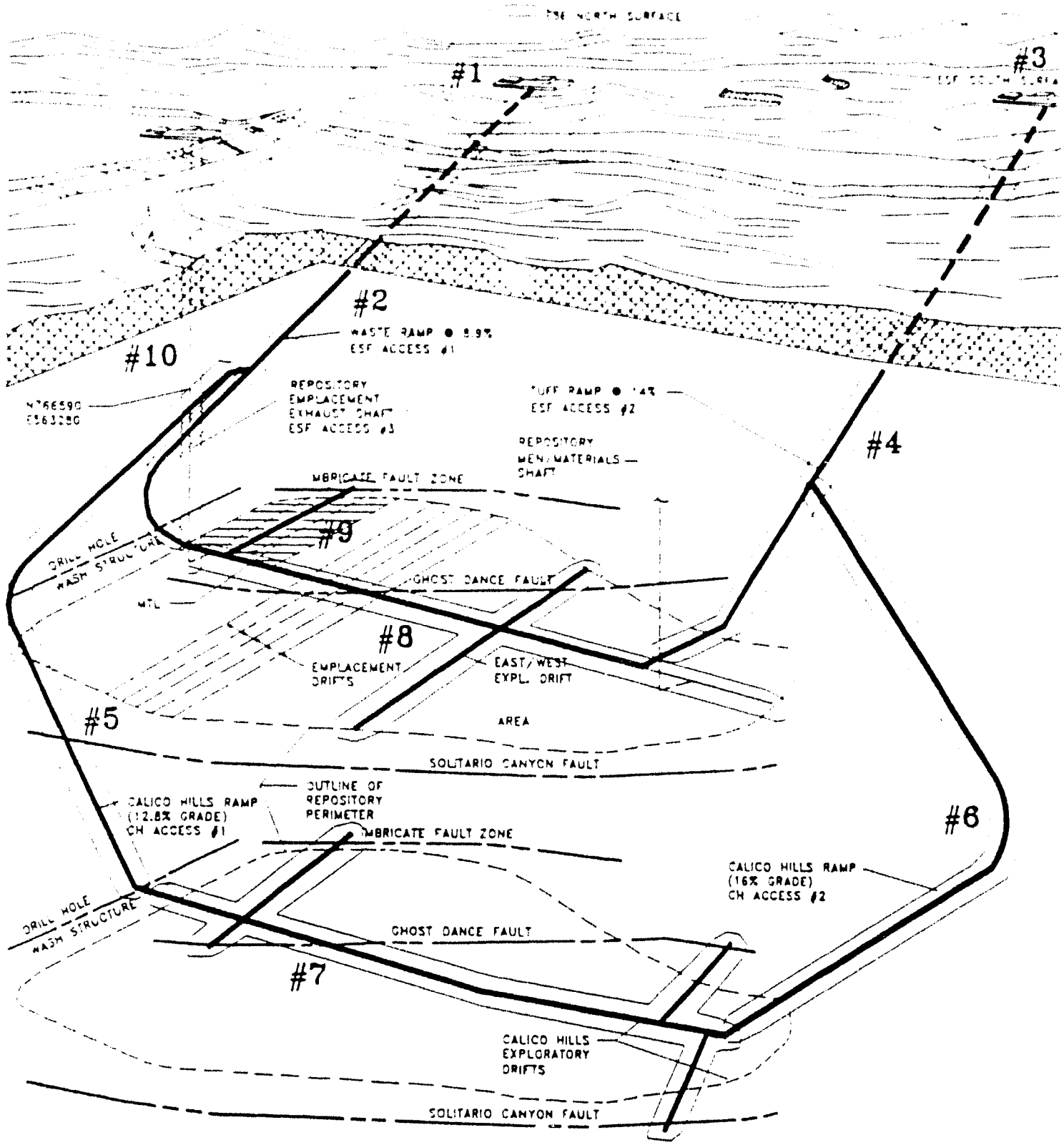
II. MANAGEMENT SYSTEM IMPROVEMENT STRATEGY (MSIS)

The MSIS is: "An aggressive effort to develop and phase in management-systems improvement as expeditously as practicable and to do so without introducing unnecessary change or perturbations in the program". This statement expresses the purpose of the MSIF effort as expressed by Dr. John Bartlett, DOE-OCRWM Director. To accomplish the above, the following points apply:

- The phase-in of the results of these initiatives into the OCRWM program will use a formal change control process;
- No program component will be removed until and unless a replacement is available;
- Existing processes will be used if they function satisfactorily.

The MSIS is:

- A new program approach to achieve DOE's mission as stated in the NWPA.
- A program to assist in implementing three initiatives stated in the Secretary of Energy's 1989 report to Congress:
  1. Ensure timely acceptance of spent fuel in Monitored Retrievable Storage Facility;
  2. Initiate scientific investigations for siting of repository;
  3. Establish improved management structure and procedures.



NOTE: THIS IS PICTORIAL ONLY AND NOT DRAWN TO SCALE

NOTE: DESIGN, CONSTRUCTION, AND TESTING PHASES SHOWN --- #2

REFERENCE DESIGN CONCEPT
FOR COMMENCING STUDY

INSERT C

PHASED DESIGN MODULES

	#30	#24	#23
1. N. RAMP FROM THE EAST TO TS (ESF ACCESS 2)	X	X	X
2. S. RAMP FROM THE EAST TO TS (TUFF RAMP S.)	X		
3. N. RAMP TO THE CH (CALICO HILLS #1 N)	X		X
4. S RAMP TO THE CH	X		
5. FULL LENGTH CH LEVEL (CH DRIFTING)	X		
CALICO HILLS PART A		X	X
CALICO HILLS PART B		X	X
6. FULL LENGTH TS LEVEL (TS DRIFTING)	X		
TS 2/3 LENGTH		X	X
7. MTL TEST AREA IN N. (MTL SINGLE DRIFT)	(X)	X	X
8. N. RAMP FROM THE N. (TUFF RAMP NORTH)			X
9. RAISED BORE SHAFT (INTERNAL TS-CH SHAFT)		X	X
10. FULL LENGTH SHAFT IN N. (ESF SURFACE SHAFT)		X	
~~11 MTL TEST AREA S. (MTL SOUTH)~~	~~X~~		

INSERT D

This effort is to be carried out "off line", thereby permitting ongoing program activities to proceed.

Enclosed is a handout detailing YMPO's part in developing the MSIS.

III. TRW CONTRACT STATUS (M&O)

The TRW contract status is as follows:

- DOE awarded TRW a management and operating contract on February 13, 1991.
- The award to TRW for the first six months is $15.8 million. The contract is for ten years.
- The TRW team includes
  - Babcock & Wilcox
  - Duke Engineering
  - Fluor Daniel
  - Intera
  - Morrison-Knudsen
  - Woodward-Clyde consultants
  - RDA
  - E. R. Johnson Associates
  - J. R. Research Associates

M&O SUPPORT TO OFFICE OF GEOLOGIC DISPOSAL

The M&O contractor shall perform the following tasks in support of the Office of Geologic Disposal:

1. Perform a study of the current integration activities and responsibilities for the Yucca Mountain Project which should include project control systems, information management systems, tracking systems, the configuration management system, etc. The results of this study should be included in the comprehensive recommendations on (1) how to make the transition with the least impact on cost and schedule, and

(1) how to reduce duplication of effort in the M&O contractor's assumption of existing or development of new systems, plans, and procedures. The recommendation should include alternatives, cost and schedule impacts, and a timeframe for implementation, including training and quality assurance requirements.

2. Review current status and perform similar studies to (1) above for the other MGDS elements, systems, waste package, site, repository, etc. and make recommendations on use of current program participants as appropriate and methods to transition to minimize impacts to the program. Recommendations should be included in comprehensive recommendation due four months after contract award.

A number of TRW personnel are at YMPO for indoctrination.

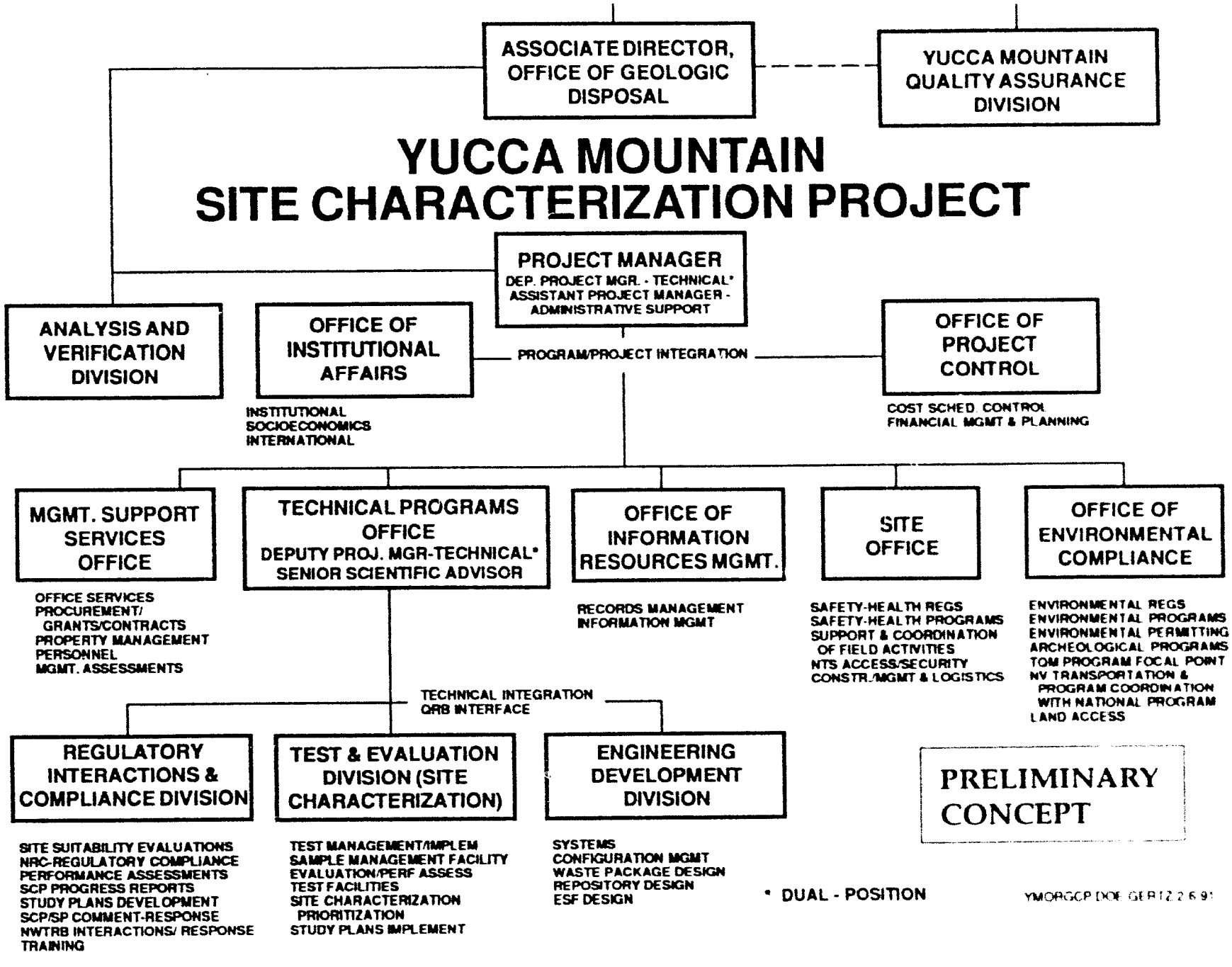
IV. PROPOSED YMPO REORGANIZATION

The inserted organization chart (Insert E) shows the proposed reorganization of the Yucca Mountain Site Characterization Project. Note the change in name. The YMPO abbreviation will still be used.

V. TECHNICAL AND MANAGEMENT DOCUMENT HIERARCHY

The technical requirements documents must be updated by March 31, 1991, to incorporate the selected ESF option. Insert F is a diagram showing portions of the Yucca Mountain document hierarchy.

WMSRD - Waste Management Systems Description
WMSR - Waste Management Systems Requirement
MGDS - Mined Geologic Disposal System
SD - Systems Description
SR - Systems Requirement


ASSOCIATE DIRECTOR, OFFICE OF GEOLOGIC DISPOSAL
YUCCA MOUNTAIN QUALITY ASSURANCE DIVISION
YUCCA MOUNTAIN SITE CHARACTERIZATION PROJECT
PROJECT MANAGER
DEP. PROJECT MGR. - TECHNICAL*
ASSISTANT PROJECT MANAGER - ADMINISTRATIVE SUPPORT
ANALYSIS AND VERIFICATION DIVISION
OFFICE OF INSTITUTIONAL AFFAIRS
INSTITUTIONAL
SOCIOECONOMICS
INTERNATIONAL
PROGRAM/PROJECT INTEGRATION
OFFICE OF PROJECT CONTROL
COST SCHED. CONTROL
FINANCIAL MGMT & PLANNING
MGMT. SUPPORT SERVICES OFFICE
OFFICE SERVICES
PROCUREMENT/ GRANTS/CONTRACTS
PROPERTY MANAGEMENT
PERSONNEL
MGMT. ASSESSMENTS
TECHNICAL PROGRAMS OFFICE
DEPUTY PROJ. MGR-TECHNICAL*
SENIOR SCIENTIFIC ADVISOR
OFFICE OF INFORMATION RESOURCES MGMT.
RECORDS MANAGEMENT
INFORMATION MGMT
SITE OFFICE
SAFETY-HEALTH REGS
SAFETY-HEALTH PROGRAMS
SUPPORT & COORDINATION OF FIELD ACTIVITIES
NTS ACCESS/SECURITY
CONSTR./MGMT & LOGISTICS
OFFICE OF ENVIRONMENTAL COMPLIANCE
ENVIRONMENTAL REGS
ENVIRONMENTAL PROGRAMS
ENVIRONMENTAL PERMITTING
ARCHEOLOGICAL PROGRAMS
TQM PROGRAM FOCAL POINT
NV TRANSPORTATION & PROGRAM COORDINATION WITH NATIONAL PROGRAM
LAND ACCESS
TECHNICAL INTEGRATION
QRB INTERFACE
REGULATORY INTERACTIONS & COMPLIANCE DIVISION
SITE SUITABILITY EVALUATIONS
NRC-REGULATORY COMPLIANCE
PERFORMANCE ASSESSMENTS
SCP PROGRESS REPORTS
STUDY PLANS DEVELOPMENT
SCP/SP COMMENT-RESPONSE
NWTRB INTERACTIONS/ RESPONSE
TRAINING
TEST & EVALUATION DIVISION (SITE CHARACTERIZATION)
TEST MANAGEMENT/IMPLEM
SAMPLE MANAGEMENT FACILITY
EVALUATION/PERF ASSESS
TEST FACILITIES
SITE CHARACTERIZATION PRIORITIZATION
STUDY PLANS IMPLEMENT
ENGINEERING DEVELOPMENT DIVISION
SYSTEMS
CONFIGURATION MGMT
WASTE PACKAGE DESIGN
REPOSITORY DESIGN
ESF DESIGN
PRELIMINARY CONCEPT
* DUAL - POSITION
YMORGCP.DOE GER 12 2 6 91

INSERT E

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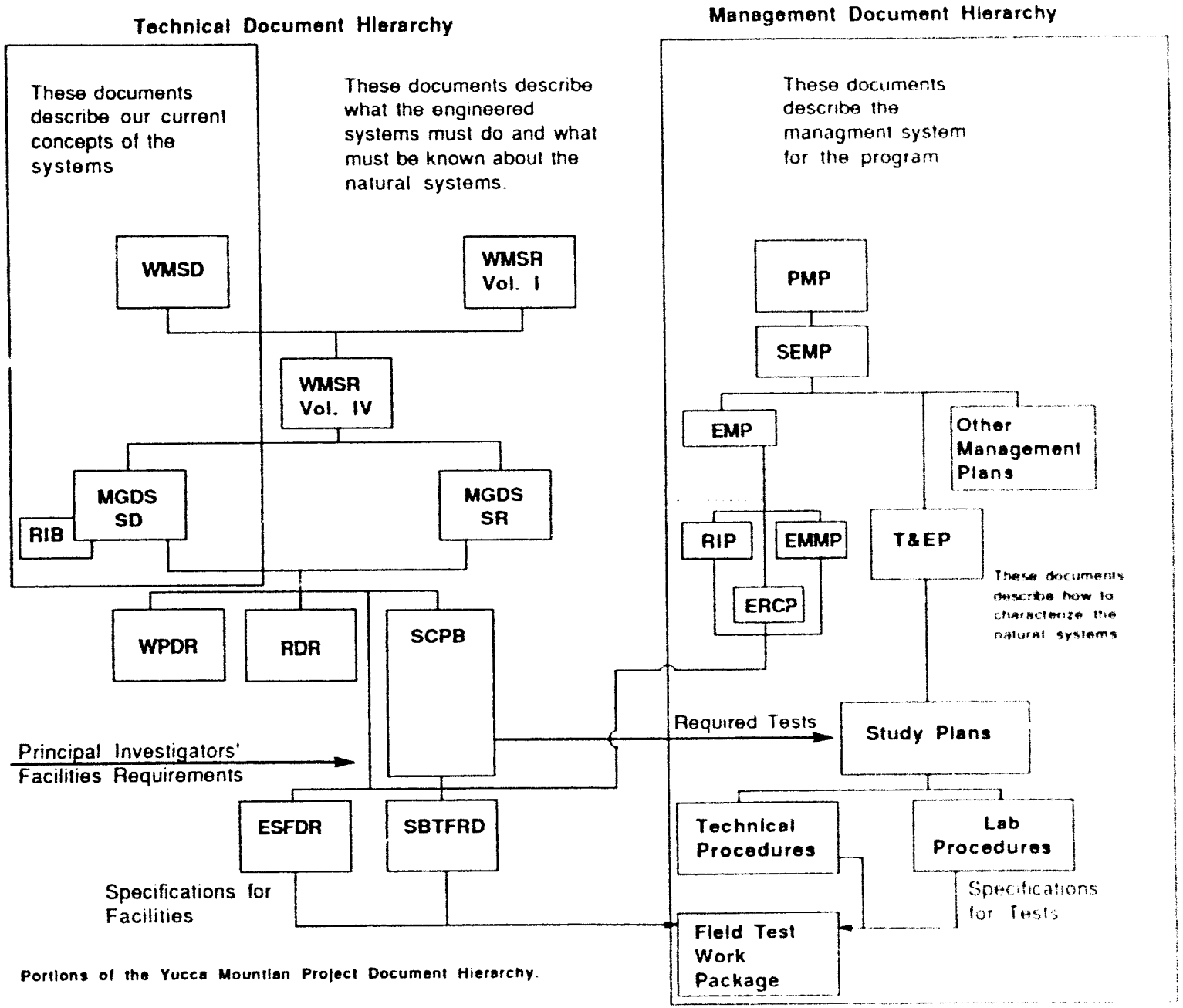


Portions of the Yucca Mountian Project Document Hierarchy.

WPDR	Waste Package Design Requirement
RDR	Repository Design Requirement
SCPB	Site Characterization Planning Basis
ESFDR	Exploratory Shaft Design Requirement
SBTFRD	Surface Based Testing Facilities Requirements Document
PMP	Project Management Plan
SEMP	Systems Engineering Management Plan
EMP	Engineering Management Plan
RIP	Retrieval Implementation Plan
EMMP	Environmental Monitoring and Mitigation Plan
T&EP	Test and Evaluation Plan
ERCP	Environmental Regulatory Compliance Plan

VI. DATA WORKSHOPS

Data workshops are planned, starting sometime in April, to explore and hopefully answer some basic questions concerning "DATA." Some of these questions are:

- What is data: what is raw data and when does raw data become enterpreted data. How is acquired data used in the licensing arena.
- When must data be made available to Yucca Mountain Participants; when to outside organizations such as the NRC or State of Nevada.
- Authority of QA, particularly auditors, in deciding which data must be QA qualified and how.
- Who has access to electronic data bases and how is legitimate access achieved.

At the invitation of Larry Hayes, USGS-TPO, I attended a meeting in Denver hosted by Mr. Hayes and including USGS principle investigators and a DOE representative. The above questions were discussed in light of the long held concerns of USGS personnel and Survey regulations. The most important single thing that came from these discussions is that almost everyone has a different understanding on every issue. It is obvious that

definitions are needed as well as an understanding of the obligations as well as the rights that each investigator has concerning data of all kinds under his or her control.

The proposed workshops will hopefully, move in the direction of gaining these understandings.

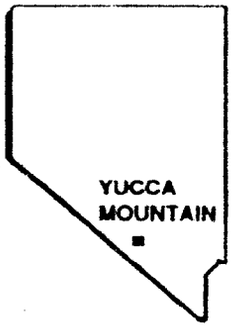
There are no new issues that this office has identified that have not been brought to management's attention.

cc w/encs.: K. Stablein, M/S 4H3; J.E. Latz
cc w/o encs.: D. Shelor, C.P. Gertz, R.E. Loux, M. Glora, G. Cook, D.M. Kunihiro, D. Weigel, B.J. Youngblood, M/S 4H3; R. Bernero, M/S 6A4; H. Thompson, M/S 17G21; H. Denton, M/S 17F2 S. Gagner, M/S 2G5; L. Kovach, M/S NLS 260; J. Martin

Enclosures: Project Participation in the Management System Improvement Strategy (MSIS), (Ted Petrie, 1/10/91); A scoping Study of Water Table Excursions Induced by Seismic and Volcanic Events, (LLNL, 11/90); Reactive Tracer Testing at the C-Well Complex, (Bruce Robinson, LANL); Net Moisture Flux in the Unsaturated Zone at Yucca Mountain --- Up, Down, or Sideways?, (D. T. Hoxie, USGS, Denver); Release of Site Characterization Program Baseline (SCPB), (Tom Bjerstedt, 2/22/91 TPO Meeting)

U.S. DEPARTMENT OF ENERGY

OCRWM



YUCCA MOUNTAIN
SITE CHARACTERIZATION
PROJECT

PROJECT PARTICIPATION IN THE MANAGEMENT SYSTEM IMPROVEMENT STRATEGY (MSIS)

PRESENTED BY

TED PETRIE

JANUARY 10, 1991

UNITED STATES DEPARTMENT OF ENERGY

MANAGEMENT SYSTEMS IMPROVEMENT STRATEGY

ISSUED BY DIRECTOR - OCRWM (J. BARTLETT)

- NEW PROGRAM APPROACH TO ACHIEVE MISSION IN NWPA
- ASSIST IN IMPLEMENTING (3) INITIATIVES OF SECRETARY'S REPORT TO CONGRESS, 1989

  1) ENSURE TIMELY ACCEPTANCE OF SPENT FUEL IN MRS

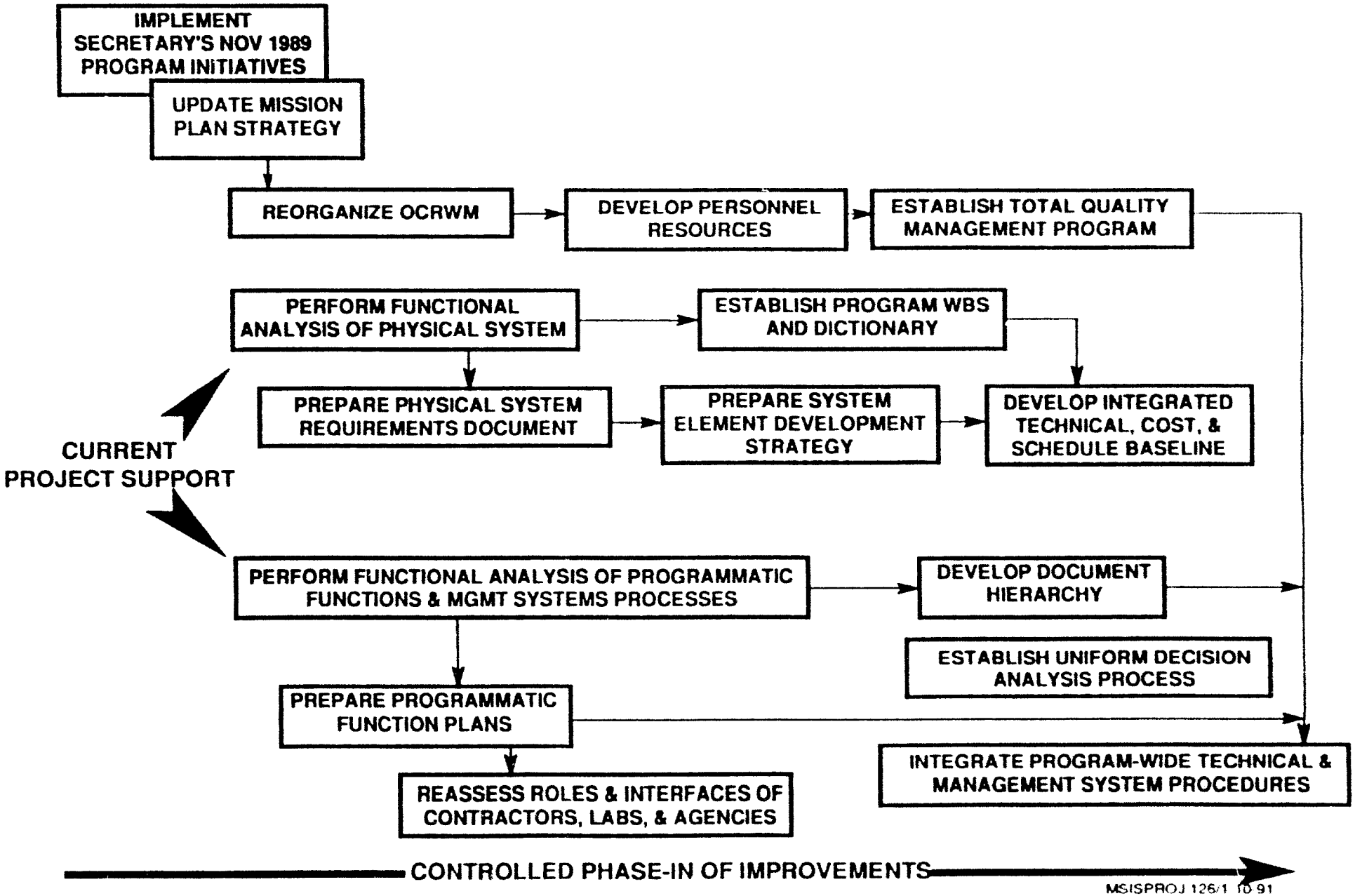
  2) INITIATE SCIENTIFIC INVESTIGATIONS FOR SITING OF REPOSITORY

  3) ESTABLISH IMPROVED MANAGEMENT STRUCTURE AND PROCEDURES

- ". . . THIS EFFORT, WHICH IS BEING CARRIED OUT 'OFF-LINE' THEREBY PERMITTING ONGOING PROGRAM ACTIVITIES TO PROCEED . . ."

MANAGEMENT SYSTEMS IMPROVEMENT STRATEGY

IMPLEMENTATION ACTIONS



MSISPROJ 126/1 10 91

MSIS FUNCTIONAL ANALYSIS

TASK:

USE SYSTEM ENGINEERING APPROACH (FUNCTIONAL ANALYSIS) TO CONDUCT COMPREHENSIVE ANALYSIS ON WASTE MANAGEMENT SYSTEM TO IDENTIFY ALL FUNCTIONS THAT MUST BE PERFORMED BY THE PROGRAMMATIC AND PHYSICAL SYSTEMS AND EACH OF ITS ELEMENTS TO ACCOMPLISH THEIR RESPECTIVE PARTS OF THE MISSION

PROJECT SUPPORT FOR MSIS FUNCTIONAL ANALYSIS

TWO TEAMS IN PLACE AND FUNCTIONING

- PROGRAMMATIC
- PHYSICAL SYSTEM
- APPROXIMATELY 30 PERSONNEL FROM PROJECT HAVE PARTICIPATED

DOE/HQ FACILITATOR FOR EACH TEAM

- DEVELOP AND DESCRIBE EACH FUNCTION
- ASSIGN REQUIREMENT
- IDENTIFY ARCHITECTURE
- PREPARE FUNCTION FLOW DIAGRAMS

PHYSICAL SYSTEM REQUIREMENTS FUNCTIONAL ANALYSIS MANAGEMENT PLAN ISSUED

PROGRAM FUNCTIONAL ANALYSIS MANAGEMENT PLAN CURRENTLY UNDER DEVELOPMENT

PHYSICAL SYSTEMS FUNCTIONAL ANALYSIS

- 5 MSIS PHYSICAL SYSTEM FUNCTIONS
- PROJECT SUPPORT - "DISPOSE WASTE" FUNCTION
- FUNCTIONAL ANALYSIS - SECOND LEVEL SUB-FUNCTIONS IN PROGRESS FOR:
  - "ISOLATE WASTE"
  - "HANDLE WASTE"
  - "EVACUATE SUBSURFACE OPENINGS"
  - "CLOSE REPOSITORY"
  - "SUPPORT DISPOSAL OPERATION"
  - "CONFIRM PERFORMANCE"

SCHEDULED COMPLETION 1/18/91

PHYSICAL SYTEMS FUNCTIONAL ANALYSIS DELIVERABLES

- COMPLETE FUNCTIONAL ANALYSIS AND PREPARE DRAFT REPORT BY 2/22/91 FOR "DISPOSE WASTE" FUNCTION
- REVIEW BY DOE/HQ SYSTEMS ENGINEERING BRANCH (SEB) IN MARCH 1991
- ISSUE REPORT ON "DISPOSE WASTE" BY 3/29/91

PROGRAMMATIC FUNCTIONAL ANALYSIS STATUS

- 15 MSIS PROGRAMMATIC FUNCTIONS TOTAL
- 3 TO BE COMPLETED AND INTERIM REPORT ISSUED FOR REVIEW BY 2/1/91
  - SYSTEMS ENGINEERING
  - DESIGN ENGINEERING
  - TEST & EVALUATION
- 2 PARTIALLY COMPLETE (IN WORK)
  - SUITABLE SITES
  - PROGRAM CONTROL
    INTERIM REPORT MAY 91

PROGRAMMATIC FUNCTIONAL ANALYSIS STATUS

(CONTINUED)

- PROGRAM FUNCTIONAL ANALYSIS MANAGEMENT PLAN
  - BEING EXPANDED TO INCLUDE IMPLEMENTATION INFORMATION
  - ISSUE BY 2/1/91 (TENTATIVE)
- SCHEDULE
  - REVISED SCHEDULE BEING DEVELOPED
  - TO BE INCLUDED IN MANAGEMENT PLAN
- FUNCTIONAL ANALYSIS CORE TEAM
  - BEING EXPANDED FROM 3 TO 6 PERSONS
  - TO MAINTAIN THE REVISED SCHEDULE

PROGRAMMATIC FUNCTIONAL ANALYSIS DELIVERABLES

INTERIM REPORTS

- DOCUMENTS THE WORK ACCOMPLISHED TO DATE
- DECISION BASIS TO AID IN VERIFICATION OF FUNCTIONS AND INTERFACES
- SCHEDULED FEBRUARY - JULY 1991 TIME FRAME

FINAL REPORT

- REFLECTS INTEGRATION OF ALL FUNCTIONS
- BASIS FOR COMPLETION OF IMPROVEMENTS TO OCRWM MANAGEMENT SYSTEMS
- PROJECTED FOR AUGUST 1991

PROGRAMMATIC FUNCTIONAL ANALYSIS DELIVERABLES

INTERIM REPORTS

- DOCUMENTS THE WORK ACCOMPLISHED TO DATE
- DECISION BASIS TO AID IN VERIFICATION OF FUNCTIONS AND INTERFACES
- SCHEDULED FEBRUARY - JULY 1991 TIME FRAME

FINAL REPORT

- REFLECTS INTEGRATION OF ALL FUNCTIONS
- BASIS FOR COMPLETION OF IMPROVEMENTS TO OCRWM MANAGEMENT SYSTEMS
- PROJECTED FOR AUGUST 1991

Chuck Carrigan
LLNL

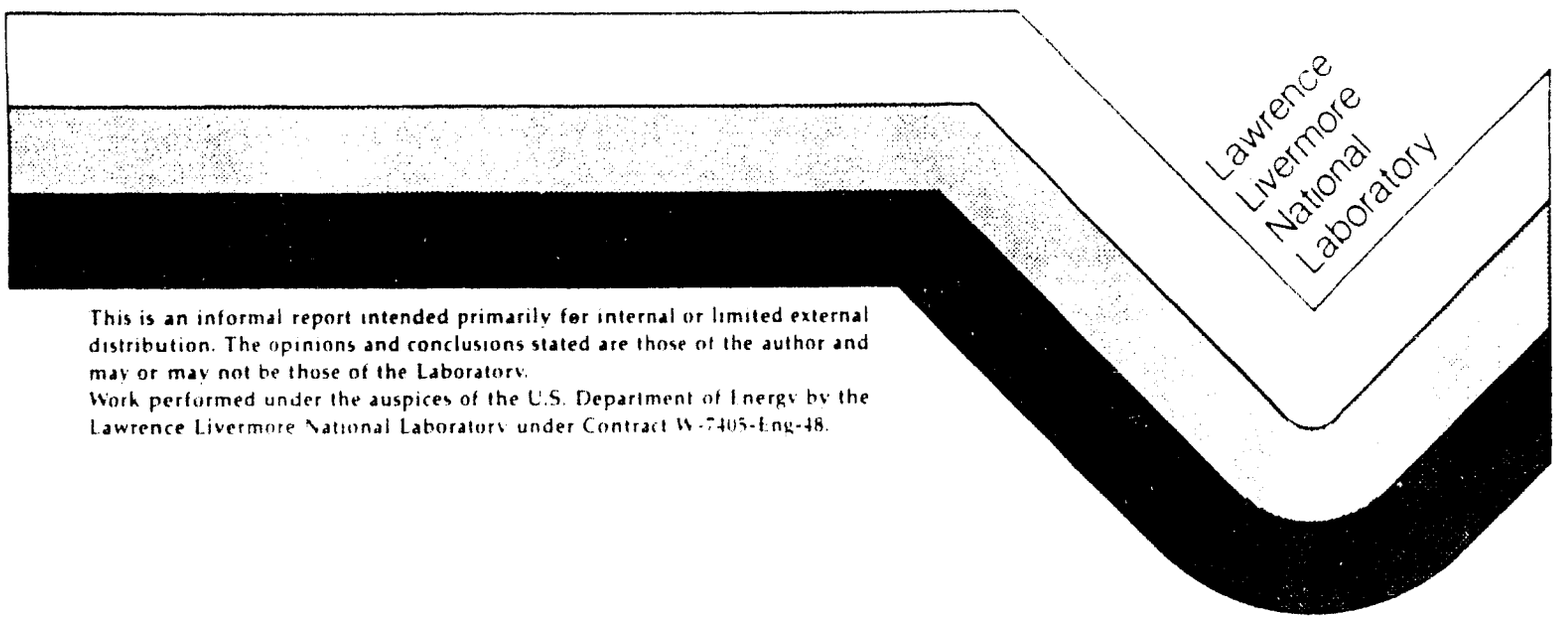
Attachment 11

UCRL-ID-105340

A Scoping Study of Water Table Excursions Induced by Seismic and Volcanic Events

Charles R. Carrigan
Geoffrey C.P. King
George E. Barr

November 1990



This is an informal report intended primarily for internal or limited external distribution. The opinions and conclusions stated are those of the author and may or may not be those of the Laboratory.
Work performed under the auspices of the U.S. Department of Energy by the Lawrence Livermore National Laboratory under Contract W-7405-Eng-48.

9103280162 910319
PDR WASTE
WM-11 PDR

Fig. 2

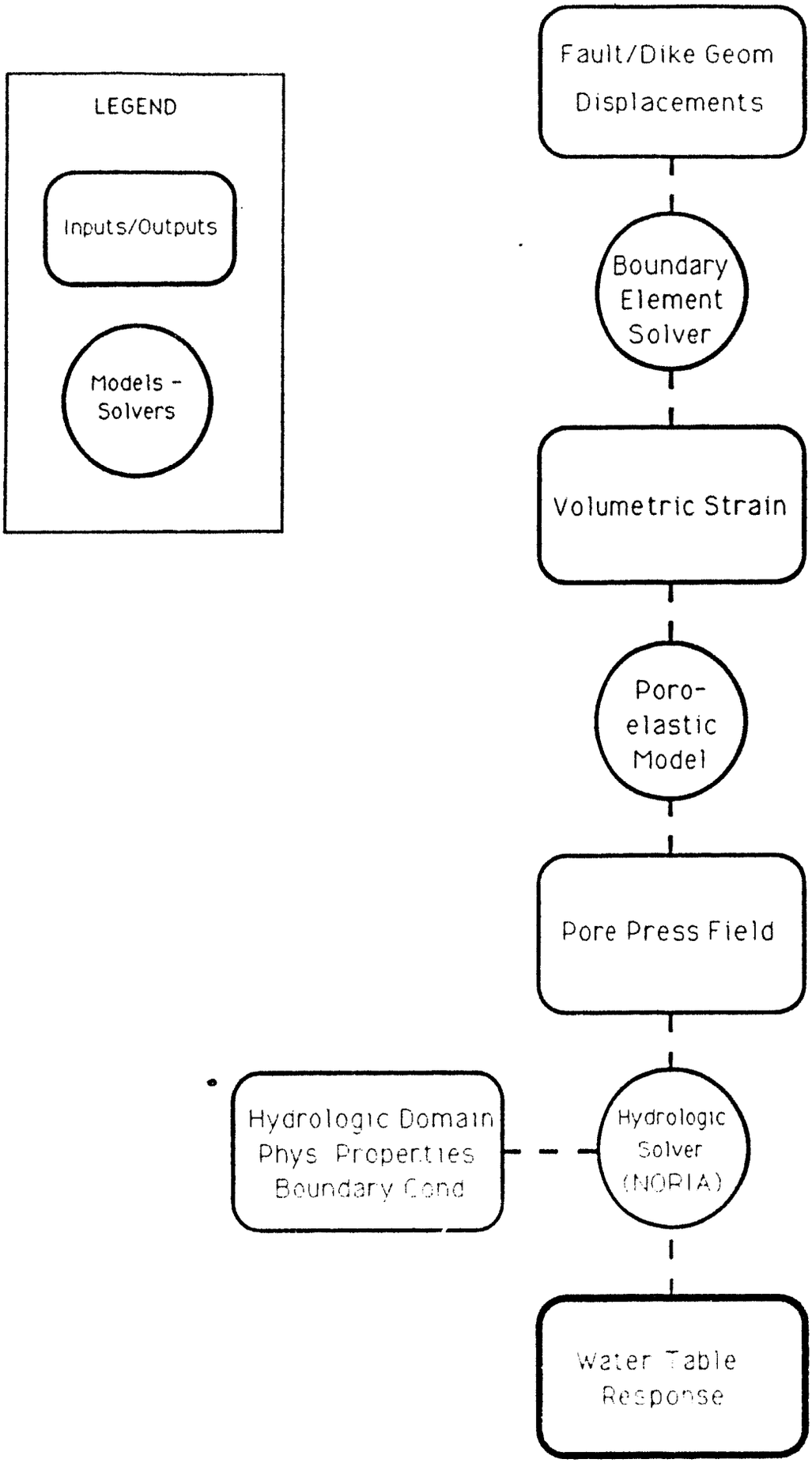
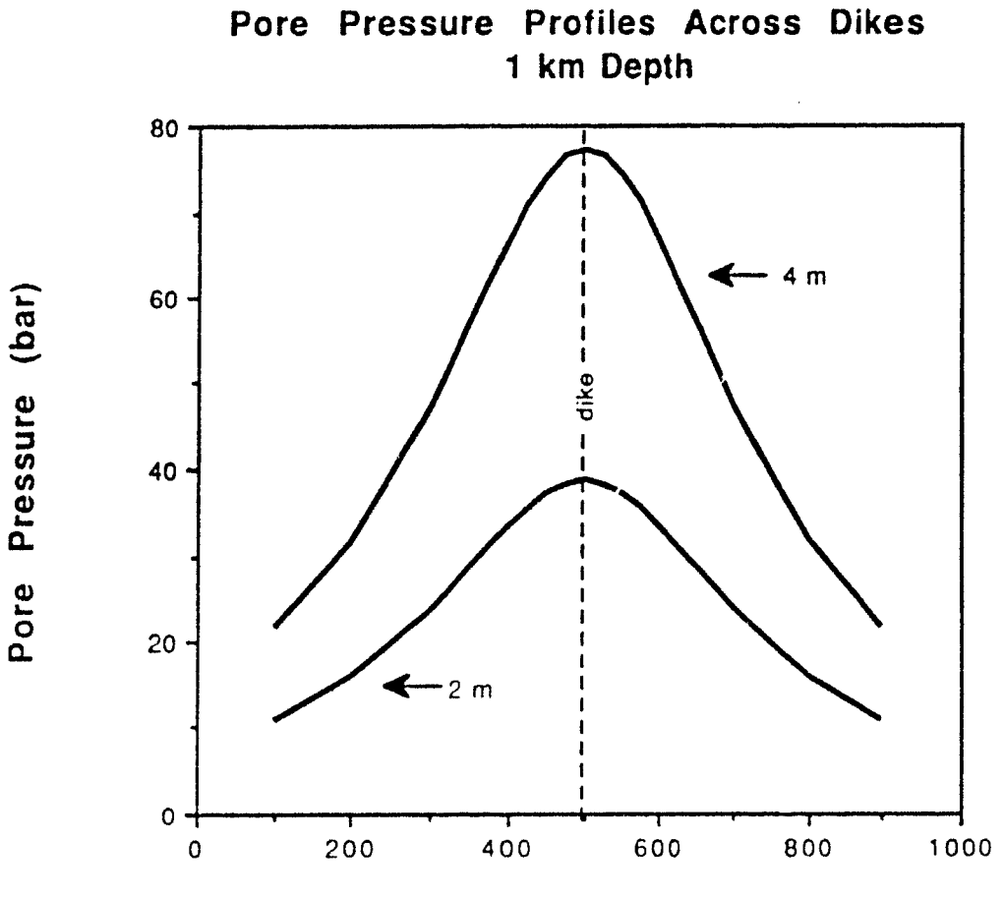
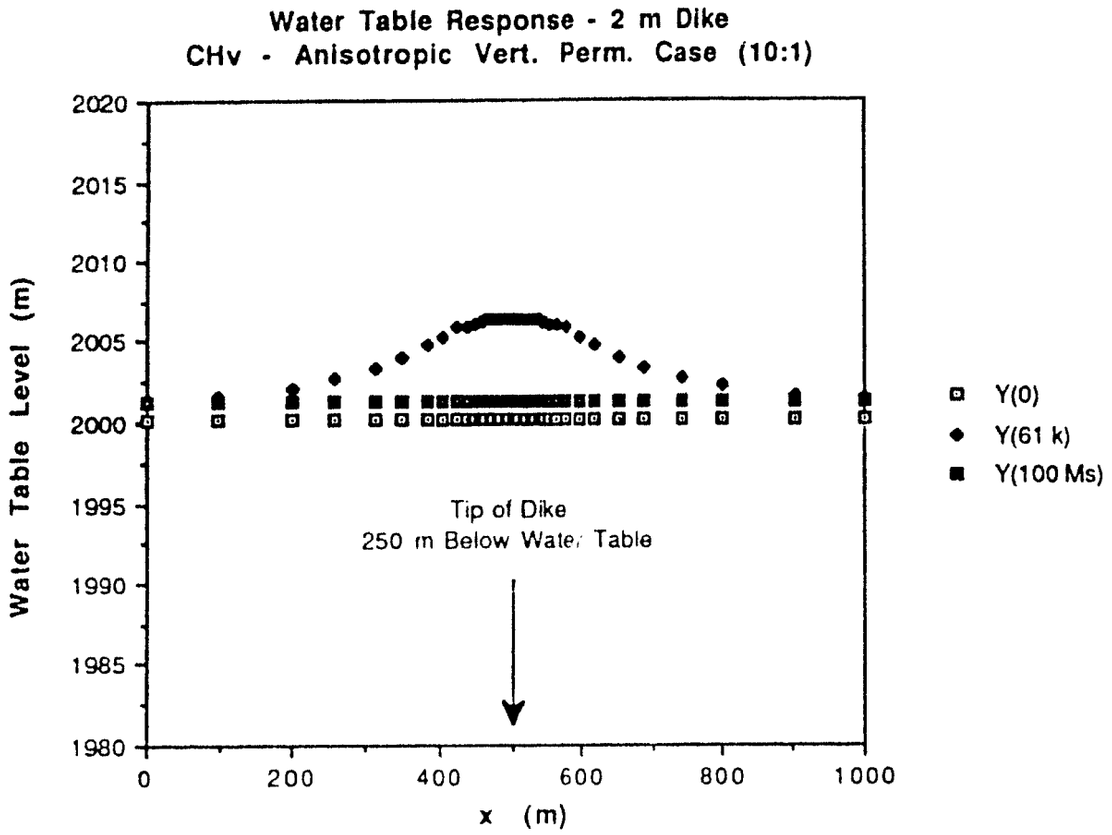

LEGEND
Inputs/Outputs
Models - Solvers
Fault/Dike Geom
Displacements
Boundary Element Solver
Volumetric Strain
Poro-elastic Model
Pore Press Field
Hydrologic Domain
Phys Properties
Boundary Cond
Hydrologic Solver (NORIA)
Water Table Response

Fig. 4


Pore Pressure Profiles Across Dikes
1 km Depth
Pore Pressure (bar)
80
60
40
20
0
0
200
400
600
800
1000
X (m)
dike
4 m
2 m



Obtained with 100 m grid over vadose zone

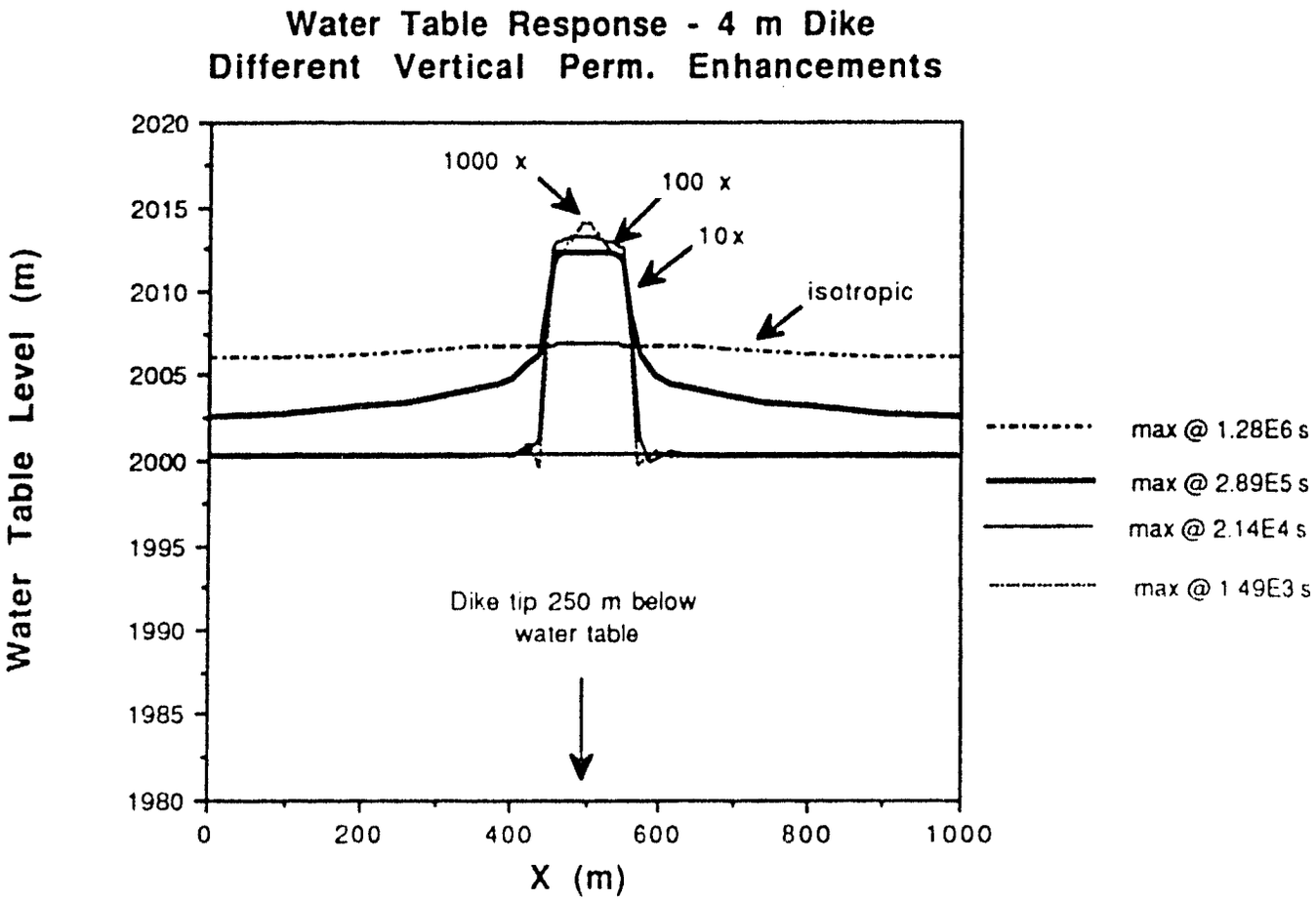
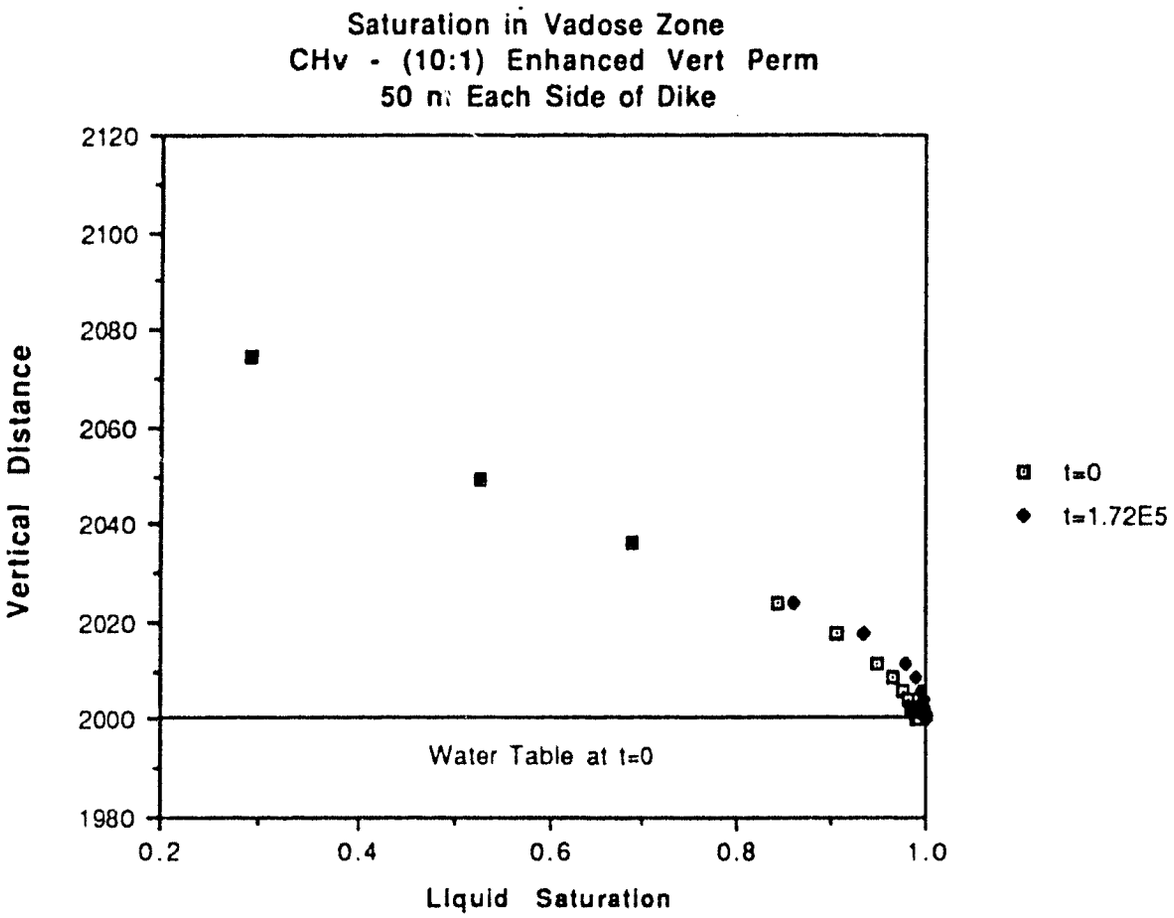
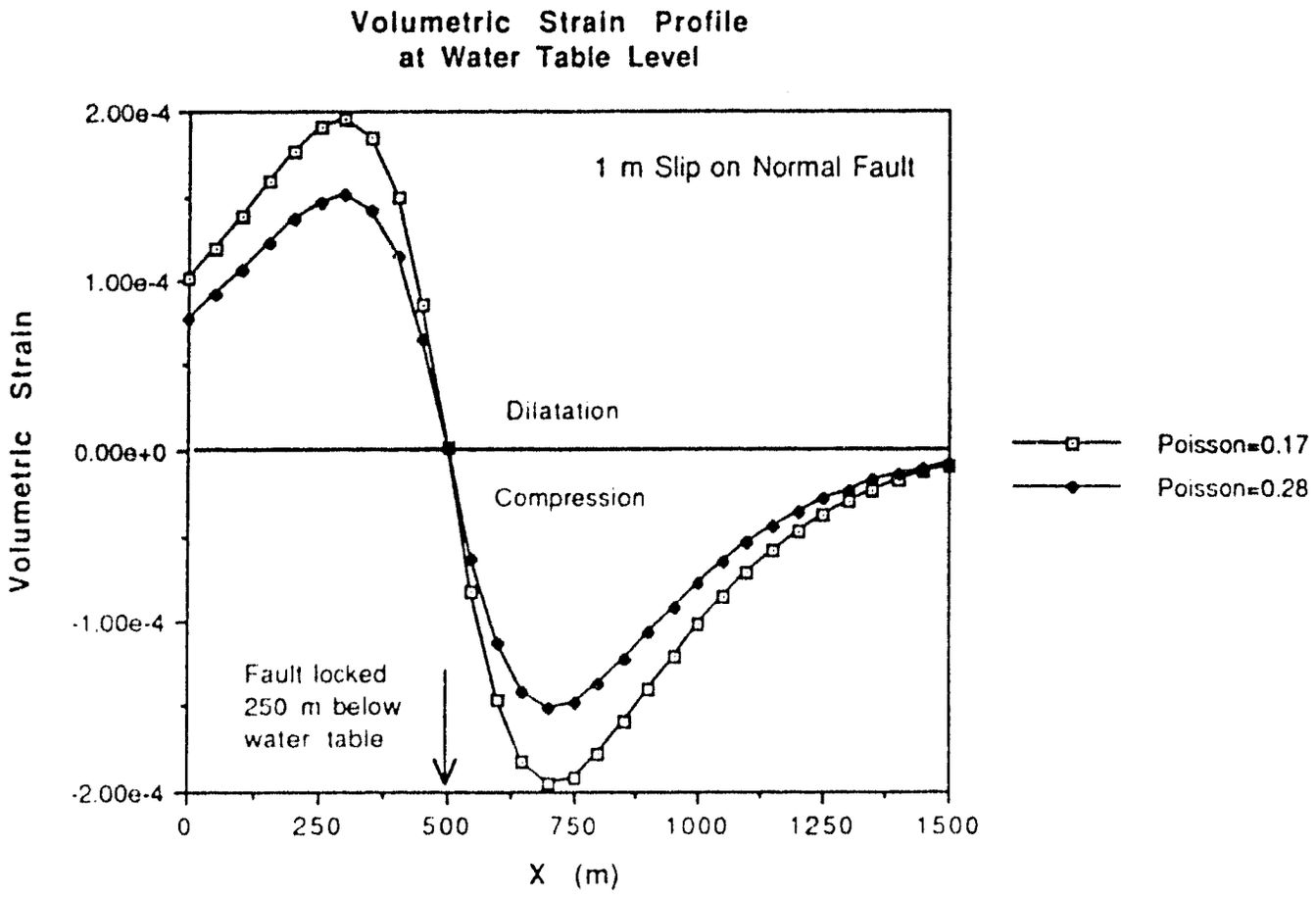
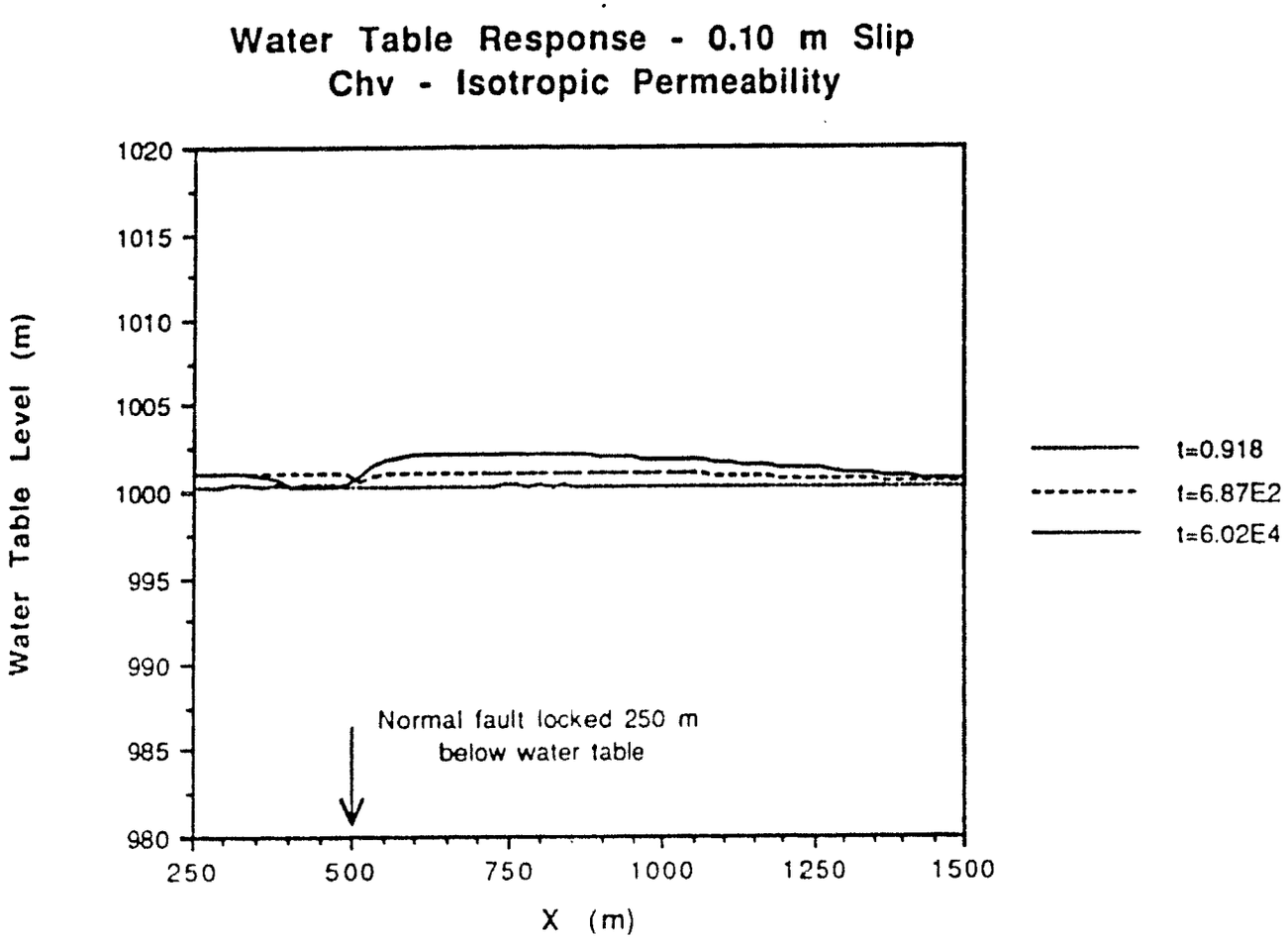
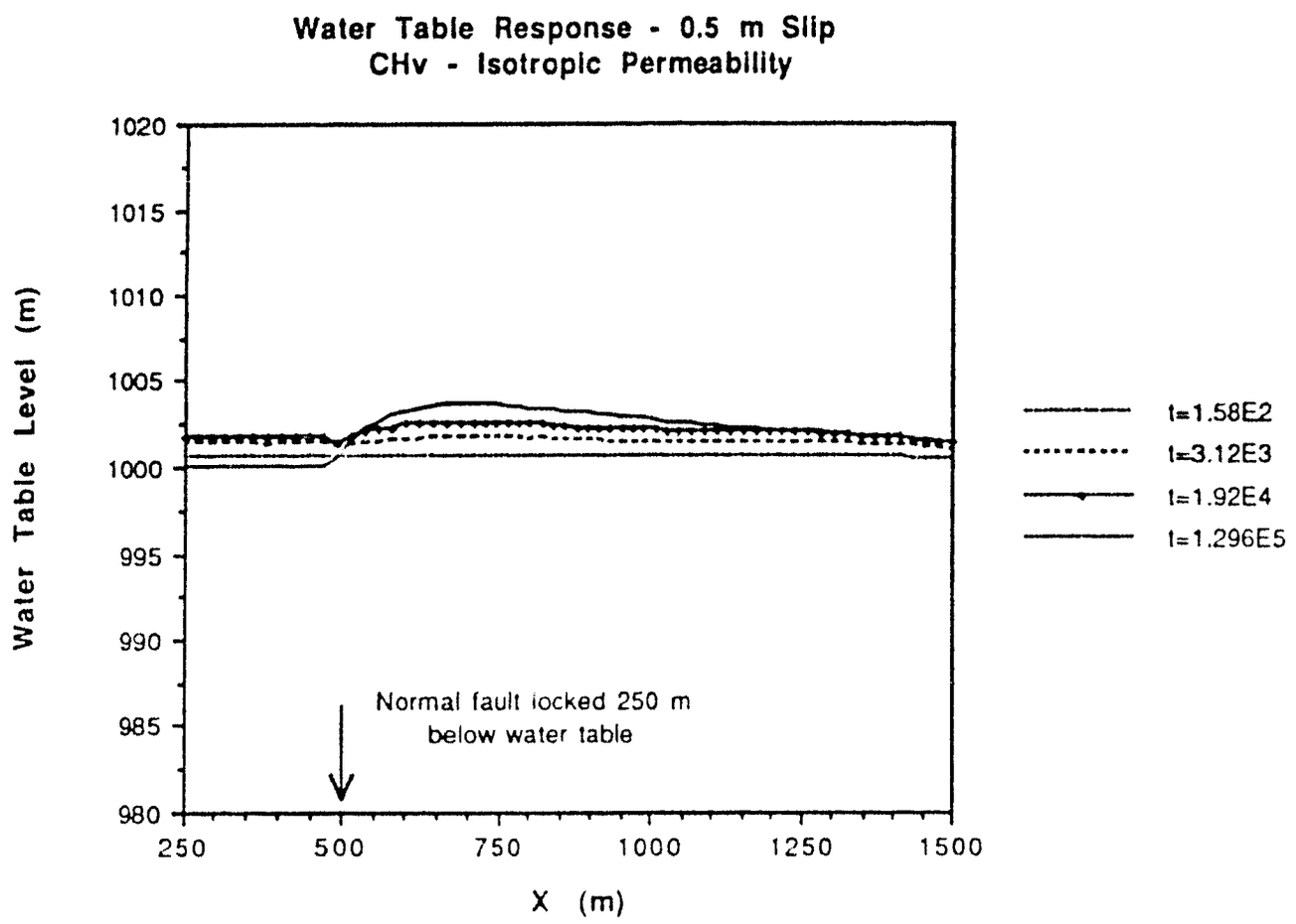

Water Table Response - 4 m Dike
Different Vertical Perm. Enhancements
Water Table Level (m)
2020
2015
2010
2005
2000
1995
1990
1985
1980
1000 x
100 x
10x
isotropic
Dike tip 250 m below
water table
0
200
400
600
800
1000
X (m)
max @ 1.28E6 s
max @ 2.89E5 s
max @ 2.14E4 s
max @ 1.49E3 s

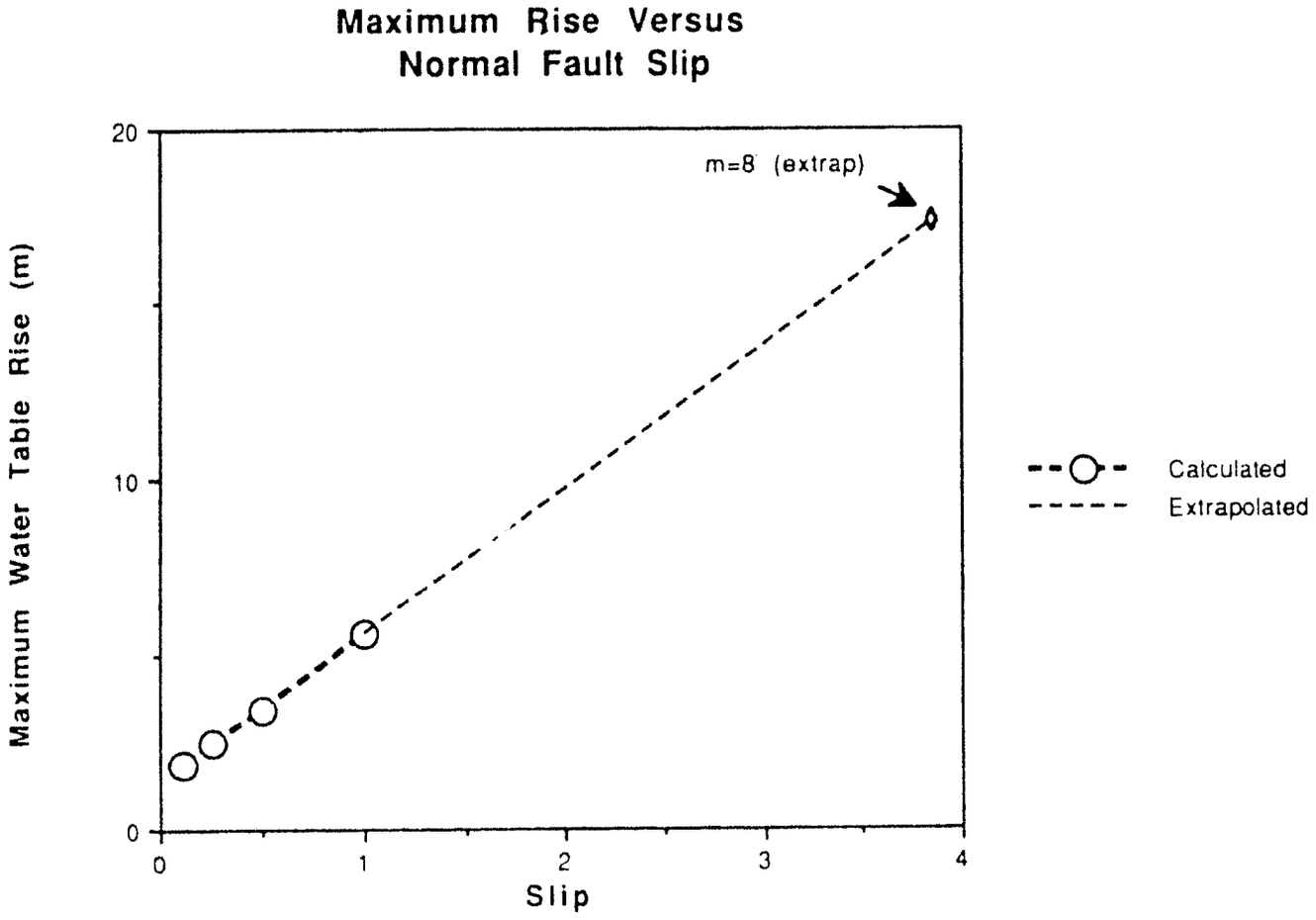
Fig. 10


Saturation in Vadose Zone
CHv - (10:1) Enhanced Vert Perm
50 m Each Side of Dike
Vertical Distance
2120
2100
2080
2060
2040
2020
2000
1980
Water Table at t=0
0.2
0.4
0.6
0.8
1.0
Liquid Saturation
t=0
t=1.72E5


Volumetric Strain Profile
at Water Table Level
1 m Slip on Normal Fault
Dilatation
Compression
Fault locked
250 m below
water table
Volumetric Strain
2.00e-4
1.00e-4
0.00e+0
-1.00e-4
-2.00e-4
0
250
500
750
1000
1250
1500
X (m)
Poisson=0.17
Poisson=0.28


Water Table Response - 0.10 m Slip
Chv - Isotropic Permeability
Water Table Level (m)
1020
1015
1010
1005
1000
995
990
985
980
250
500
750
1000
1250
1500
X (m)
Normal fault locked 250 m
below water table
t=0.918
t=6.87E2
t=6.02E4


Water Table Response - 0.5 m Slip
CHv - Isotropic Permeability
Water Table Level (m)
1020
1015
1010
1005
1000
995
990
985
980
250
500
750
1000
1250
1500
X (m)
Normal fault locked 250 m
below water table
t=1.58E2
t=3.12E3
t=1.92E4
t=1.296E5


Maximum Rise Versus
Normal Fault Slip
Maximum Water Table Rise (m)
20
10
0
m=8 (extrap)
0
1
2
3
4
Slip
Calculated
Extrapolated

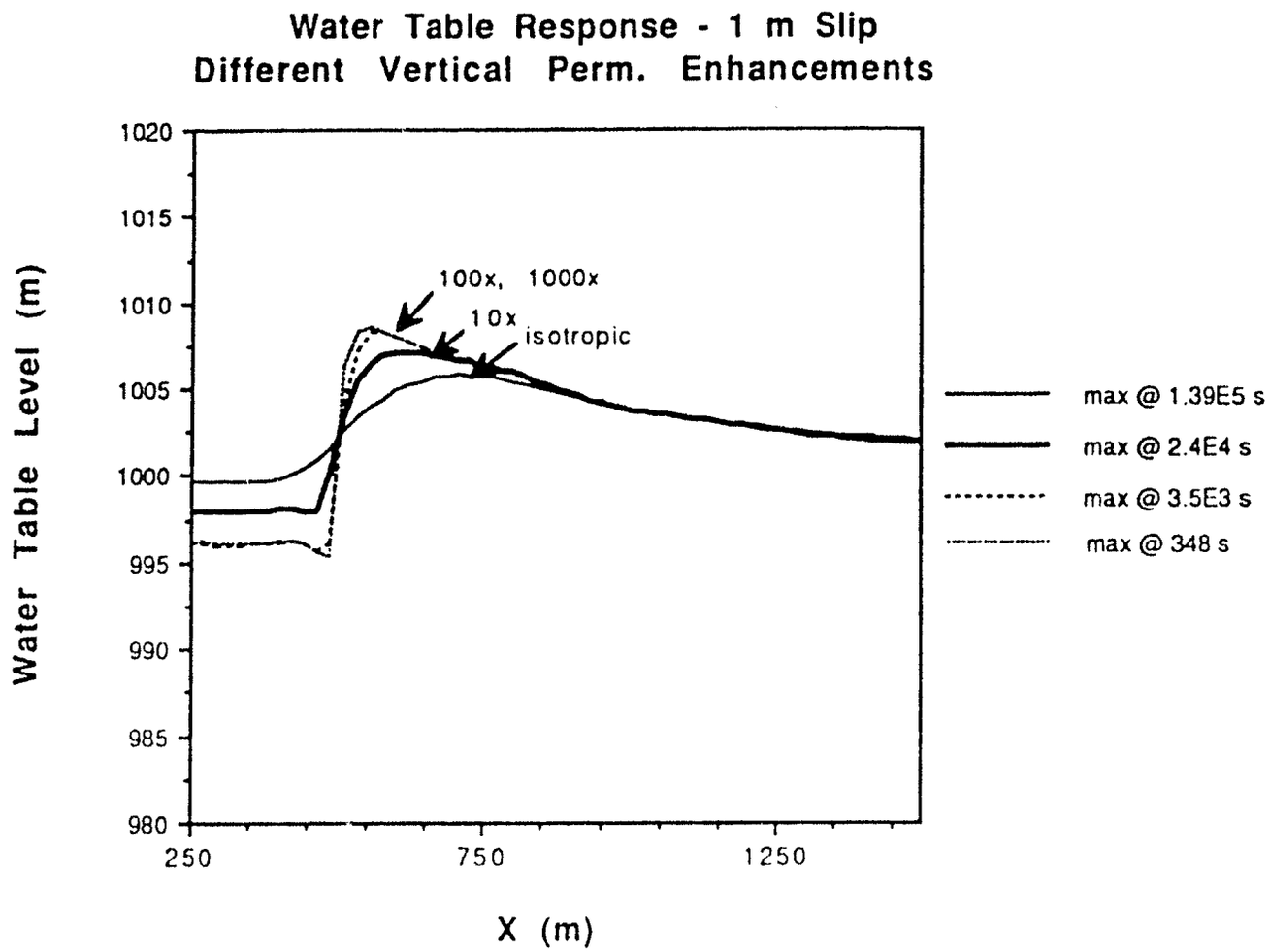
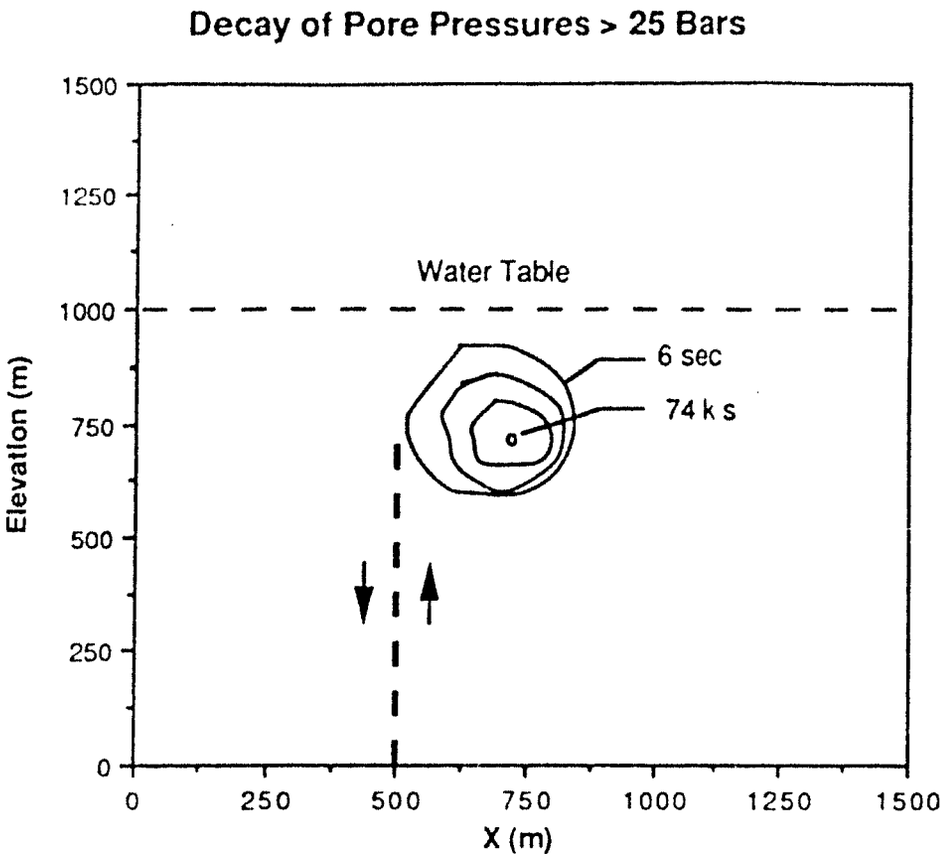
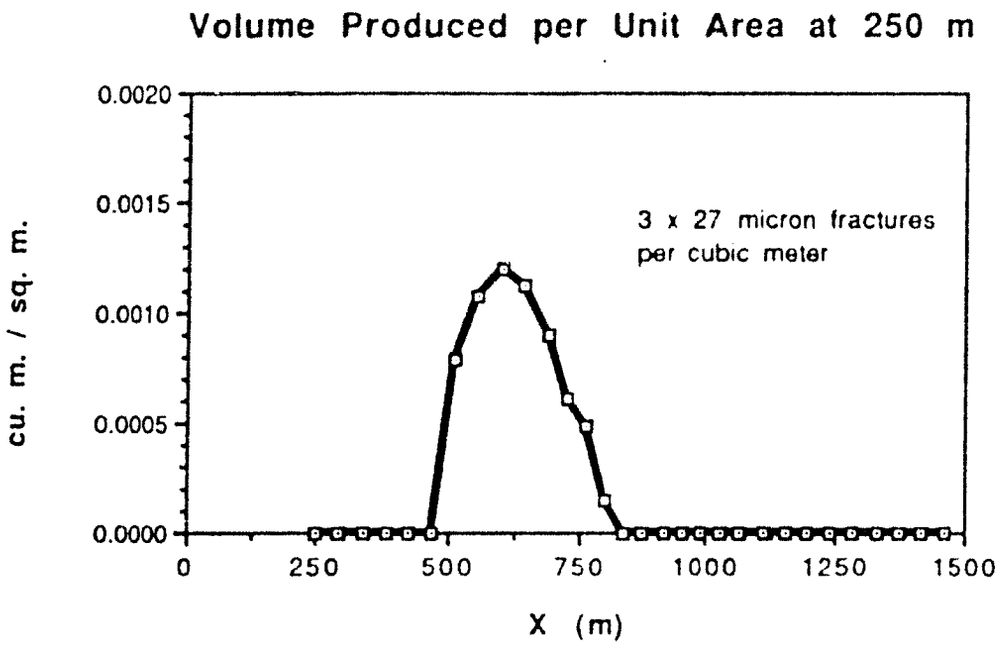
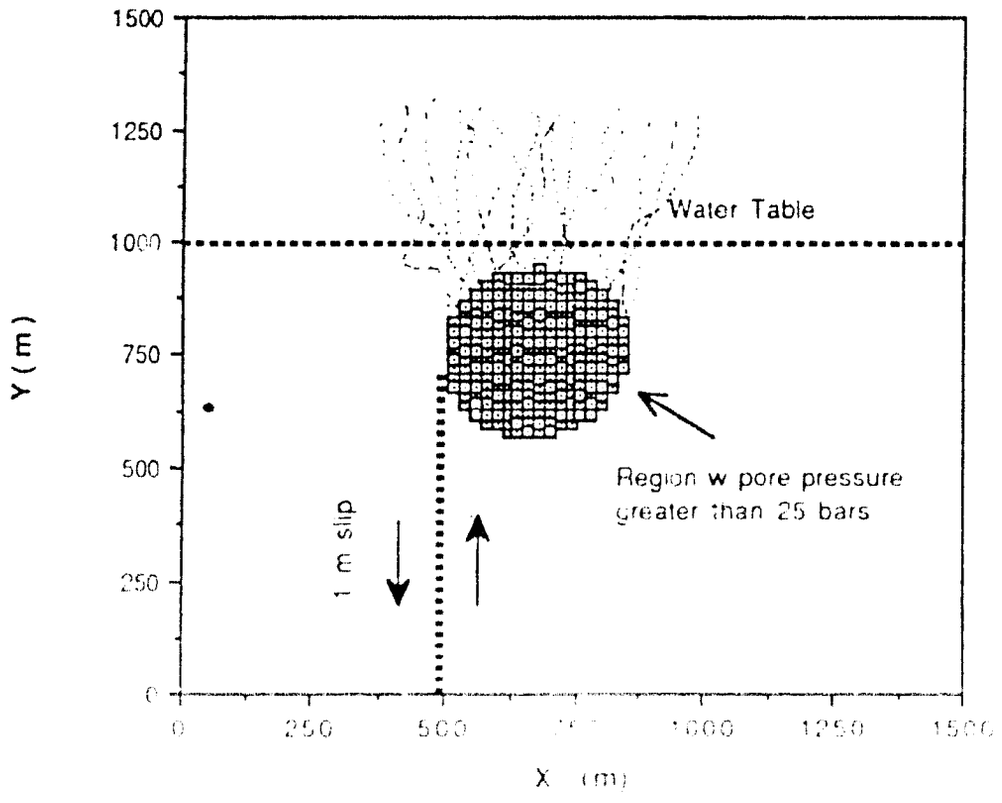
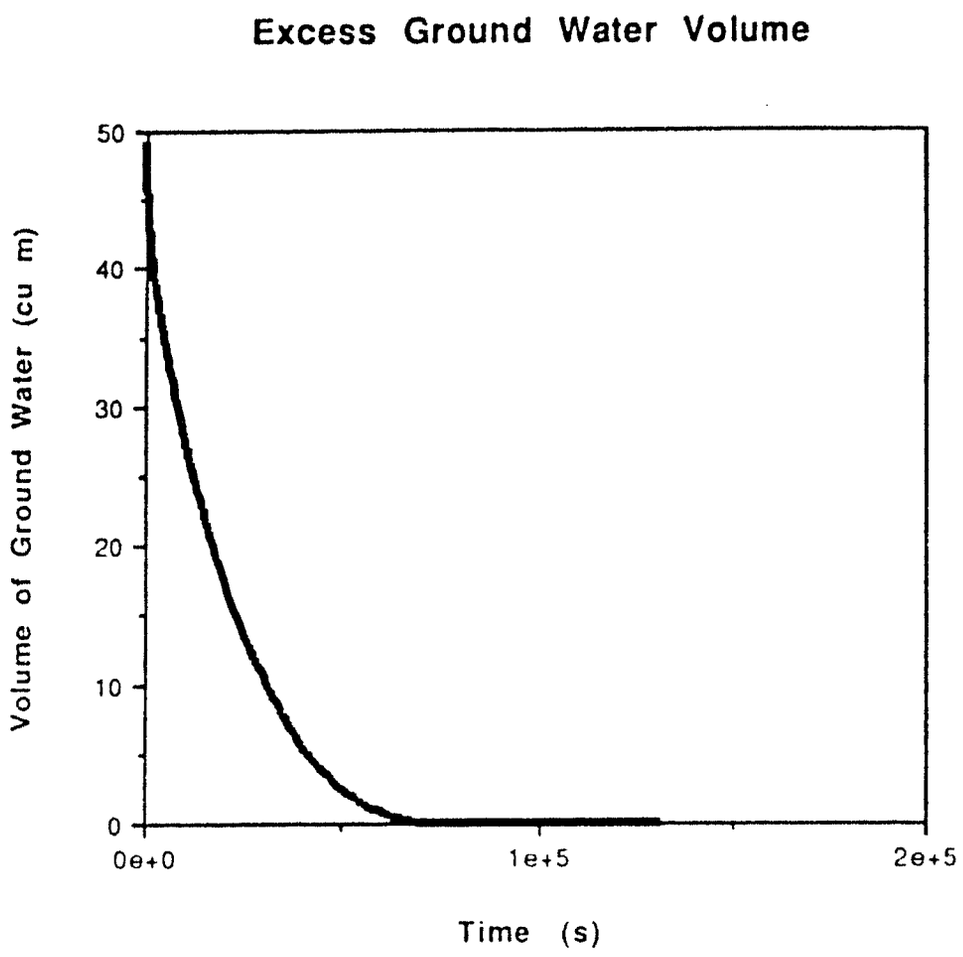

Water Table Response - 1 m Slip
Different Vertical Perm. Enhancements
Water Table Level (m)
1020
1015
1010
1005
1000
995
990
985
980
250
750
1250
X (m)
100x, 1000x
10x
isotropic
max @ 1.39E5 s
max @ 2.4E4 s
max @ 3.5E3 s
max @ 348 s

Fig. 22


Decay of Pore Pressures > 25 Bars
1500
1250
1000
750
500
250
0
Elevation (m)
Water Table
6 sec
74 k s
0
250
500
750
1000
1250
1500
X (m)


Volume Produced per Unit Area at 250 m
cu. m. / sq. m.
0.0020
0.0015
0.0010
0.0005
0.0000
3 x 27 micron fractures
per cubic meter
0
250
500
750
1000
1250
1500
X (m)


Y (m)
1500
1250
1000
750
500
250
0
Water Table
Region w pore pressure
greater than 25 bars
1 m slip
0
250
500
1000
1250
1500
X (m)


Excess Ground Water Volume
Volume of Ground Water (cu m)
50
40
30
20
10
0
0e+0
1e+5
2e+5
Time (s)

B. Robinson
Card
Attachment 10

REACTIVE TRACER TESTING AT THE C-WELL COMPLEX

Bruce Robinson
Los Alamos National Laboratory

Objectives of the Study

- Demonstrate applicability of laboratory-determined sorption parameters
- Validate transport model for saturated zone
- Evaluate potential for colloid transport

Approach

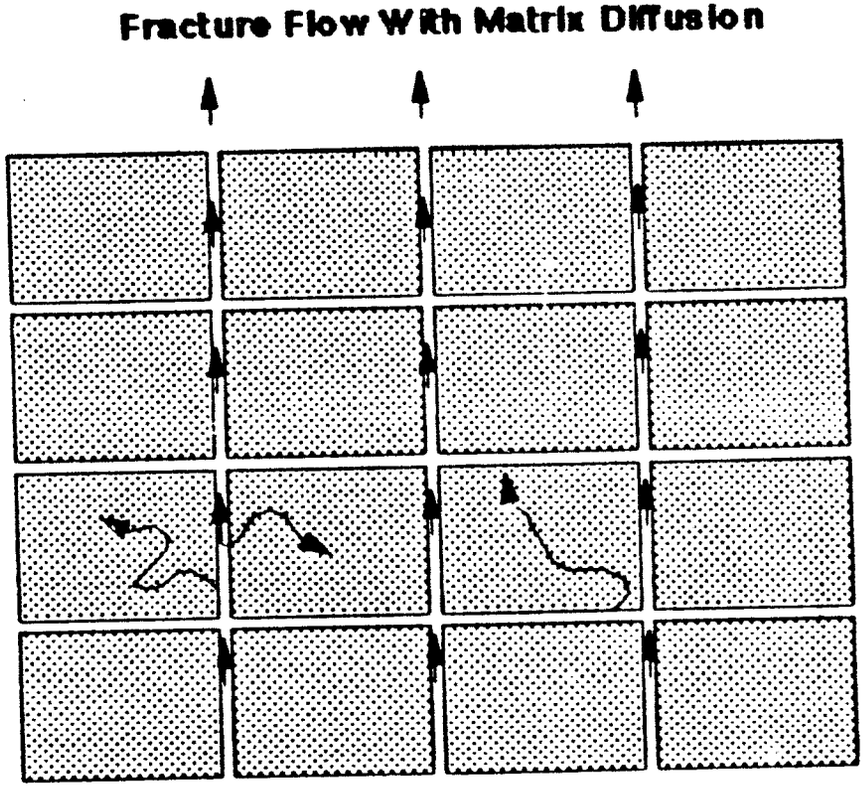
Test transport models at the field scale using carefully selected, well-characterized tracers.

- Radionuclide sorption: LiBr
- Colloid transport: polystyrene microspheres

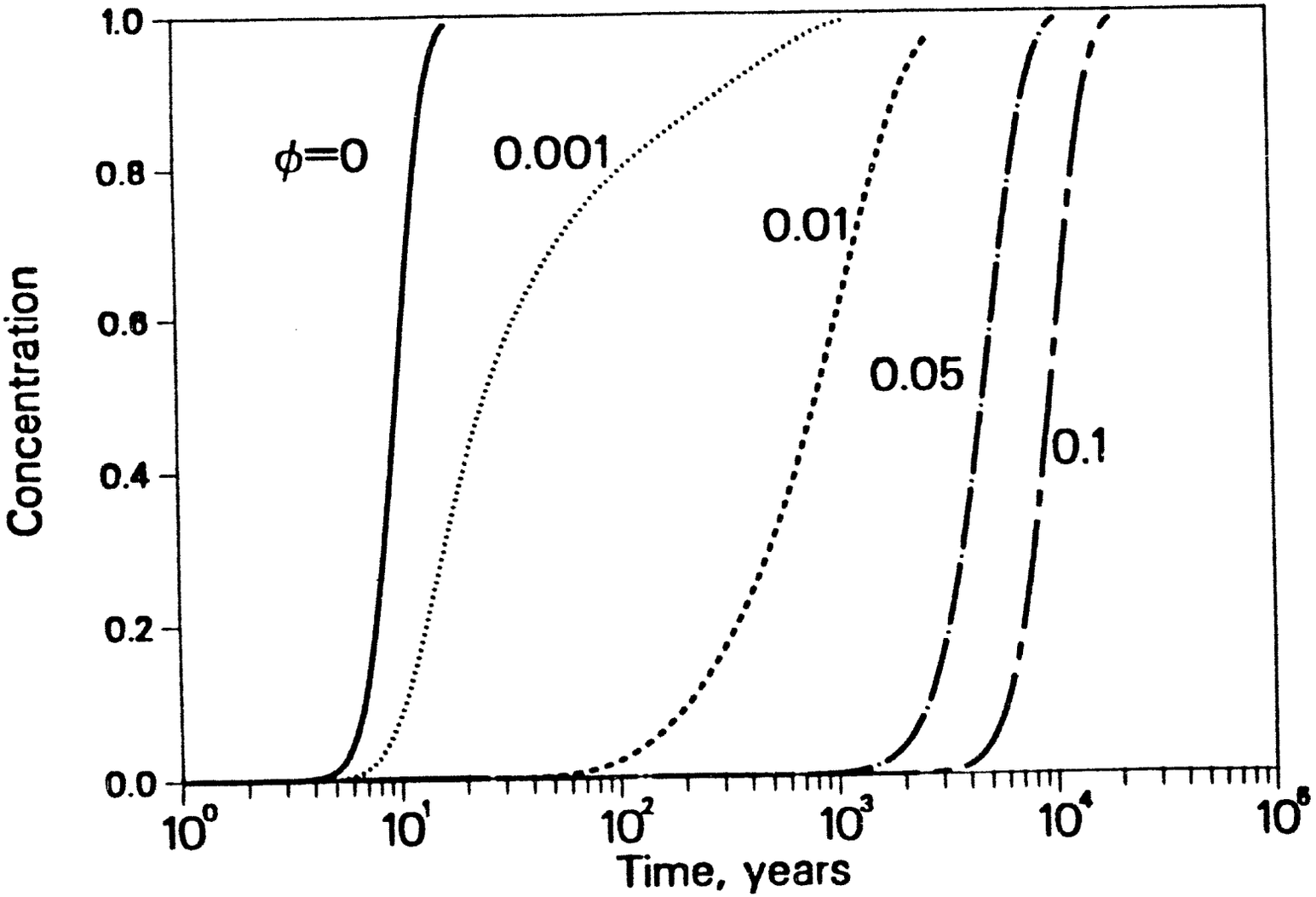
Test Procedure

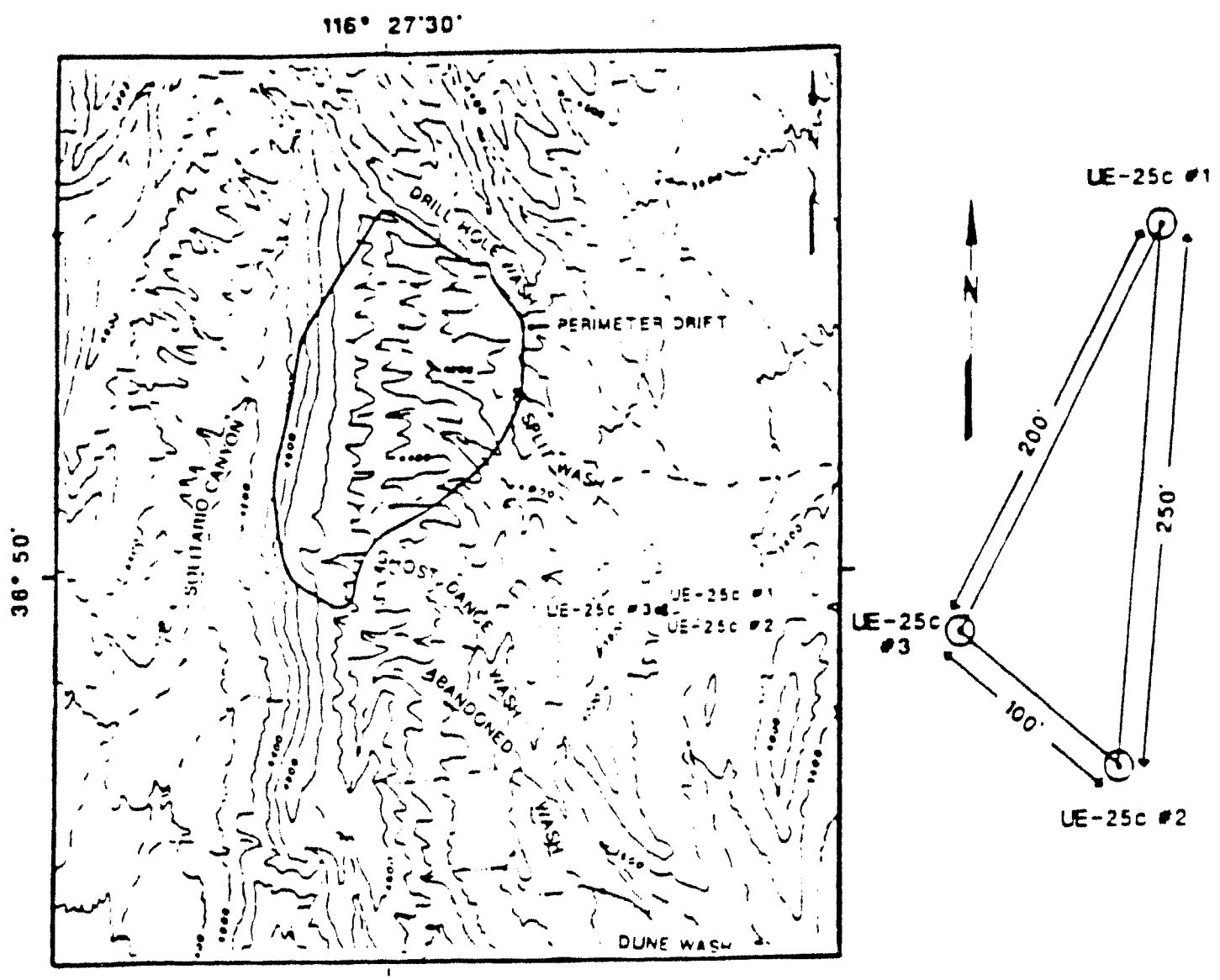
- Identify and characterize tracers in the laboratory
- Conduct field tests
- Simulate hydrologic and conservative tracer behavior
- Determine the applicability of laboratory sorption data for predicting field behavior

Fracture Flow With Matrix Diffusion

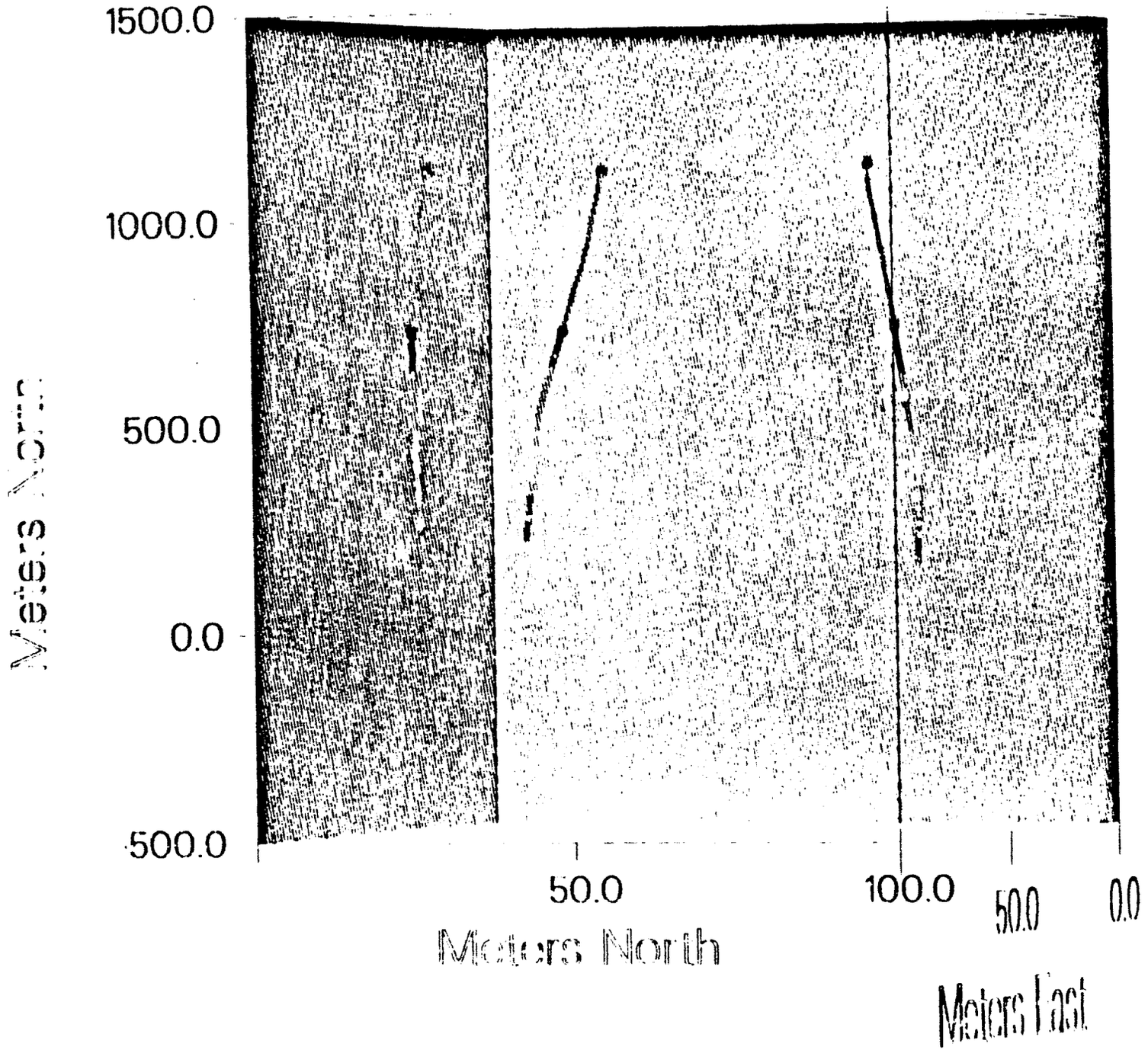


Effect of Matrix Porosity

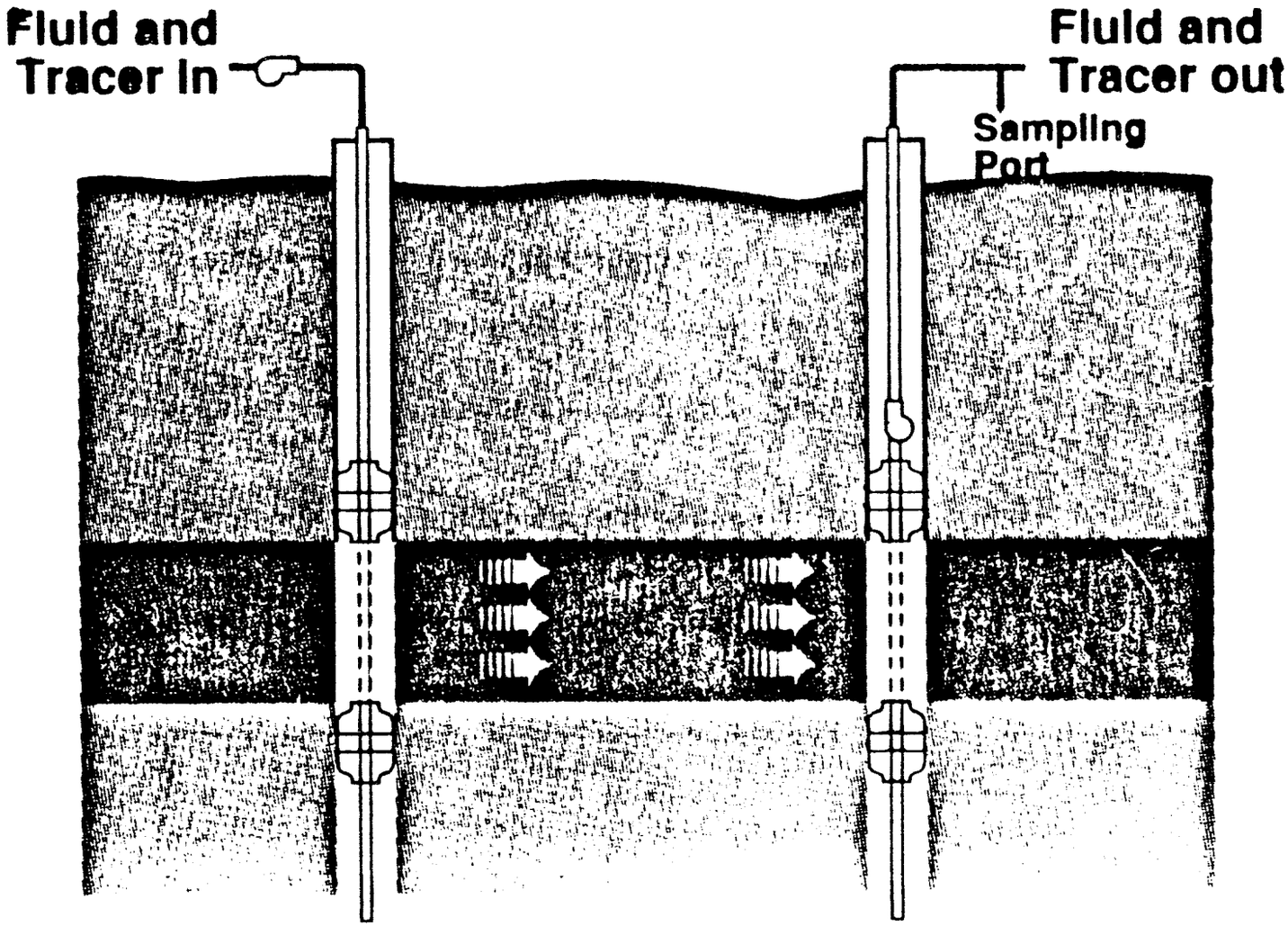


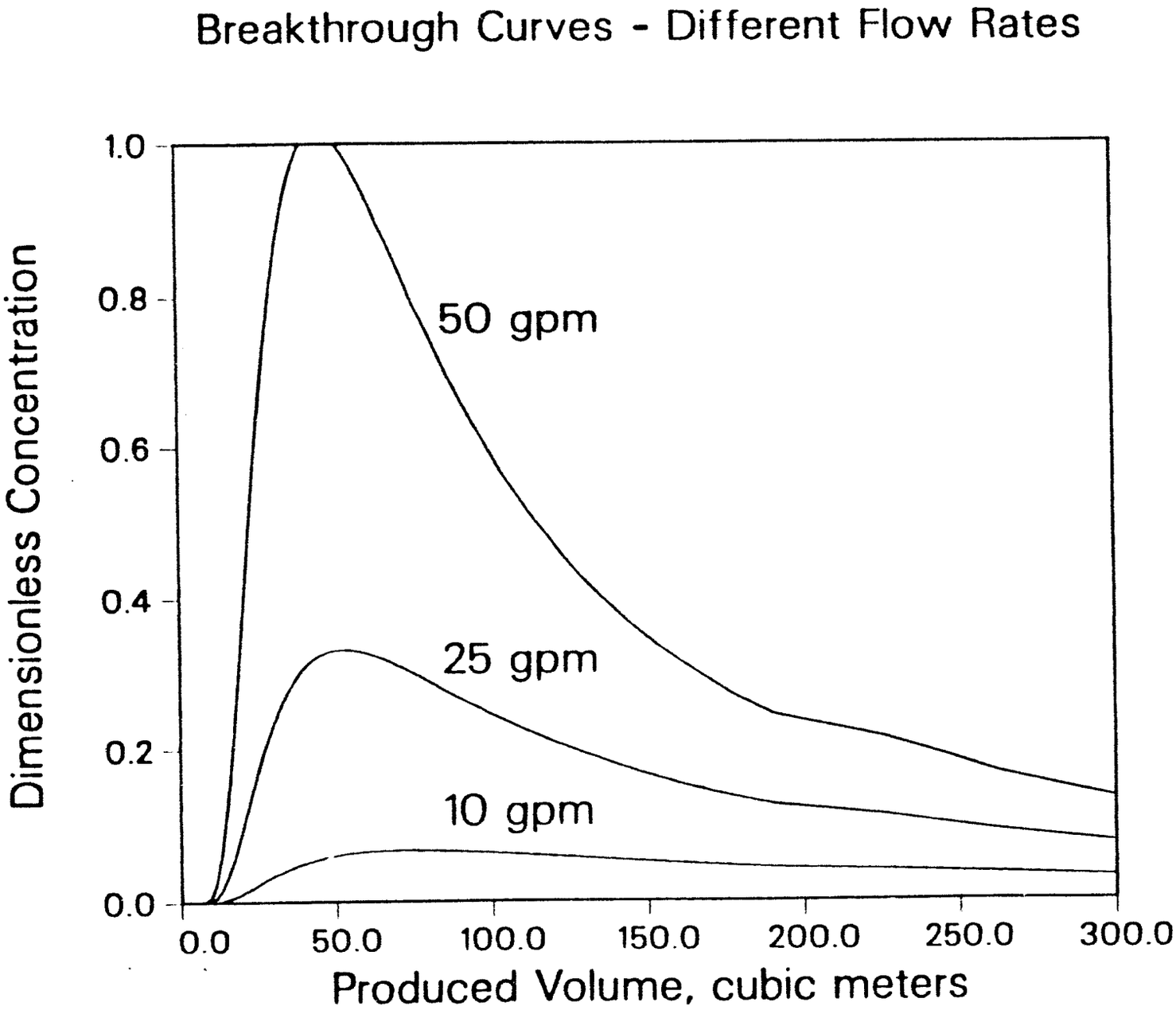

116° 27'30"
36° 50'
DRILL HOLE WASH
PERIMETER DRIFT
SOLITARIO CANYON
SOLITAIRE WASH
GHOST DANCE FAULT
ABANDONED WASH
DUNE WASH
UE-25c #1
UE-25c #3
UE-25c #2
1,000 0 3,000 7,000 FEET
1 .5 0 1 KILOMETER
N
200'
250'
100'
UE-25c #1
UE-25c #3
UE-25c #2

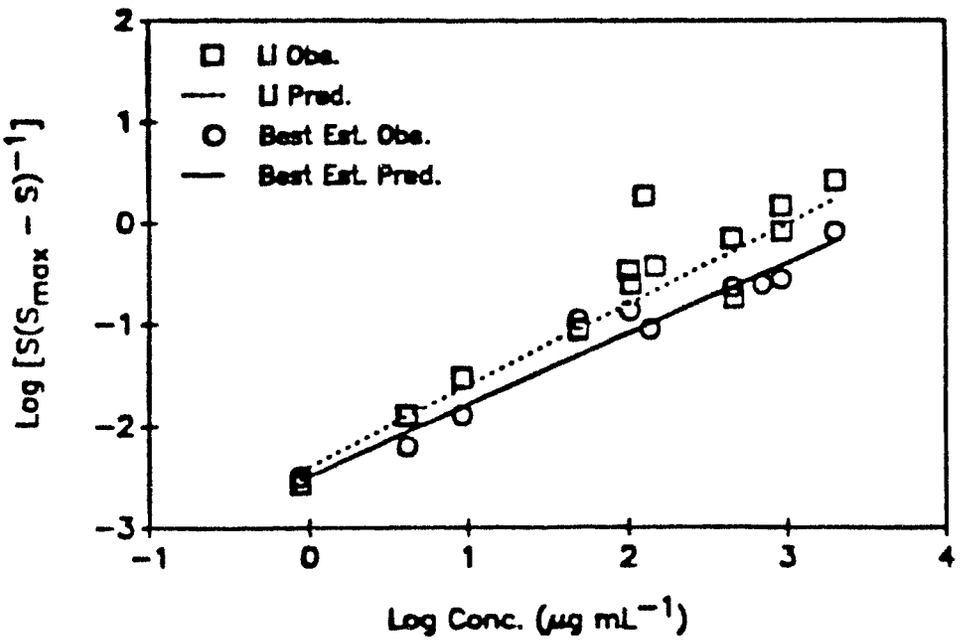
PRINT THIS SIDE

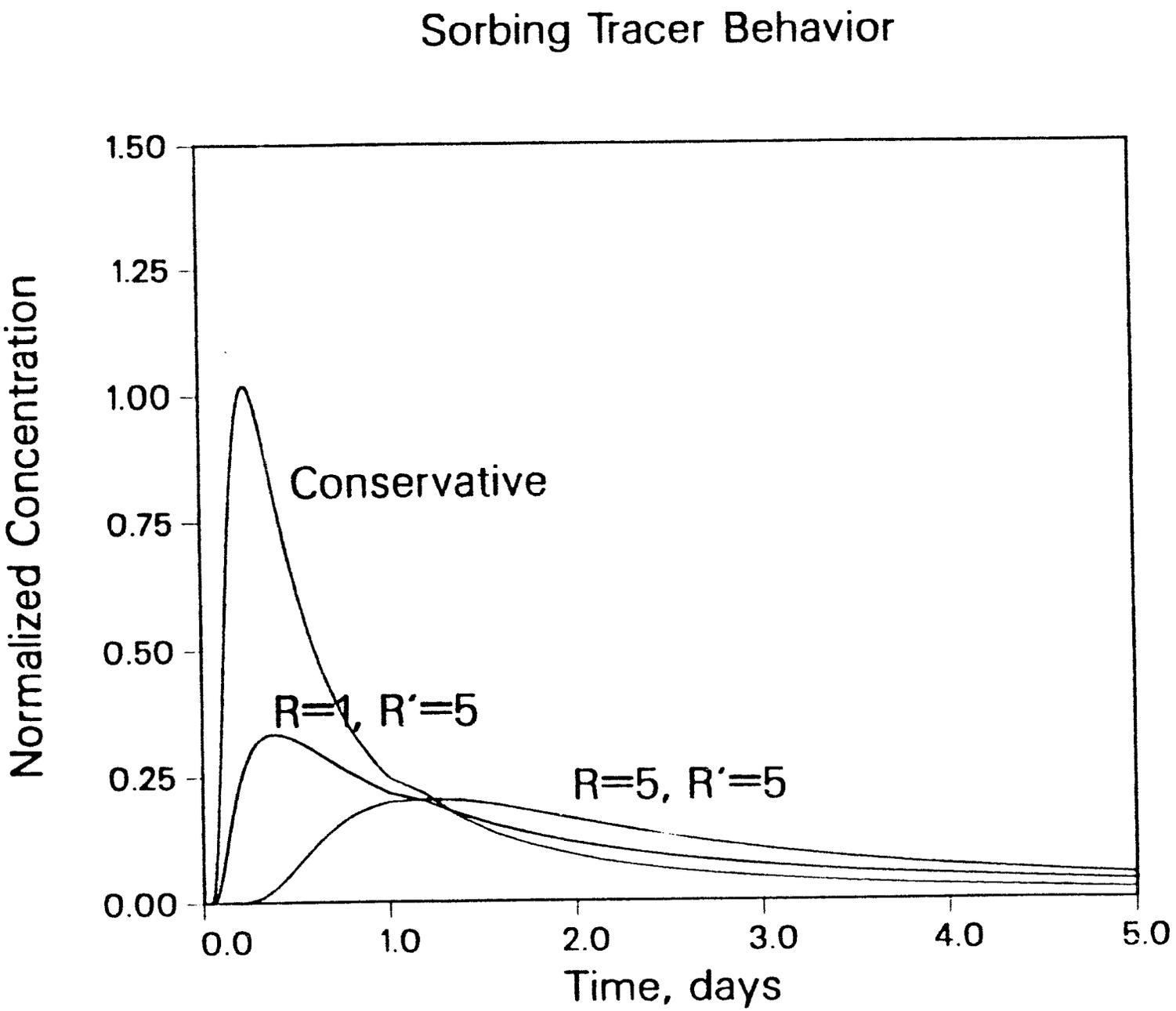

1500.0
1000.0
500.0
0.0
-500.0
Meters North
50.0
100.0
50.0
0.0
Meters North
Meters East

Two Well Recirculating Tracer Test

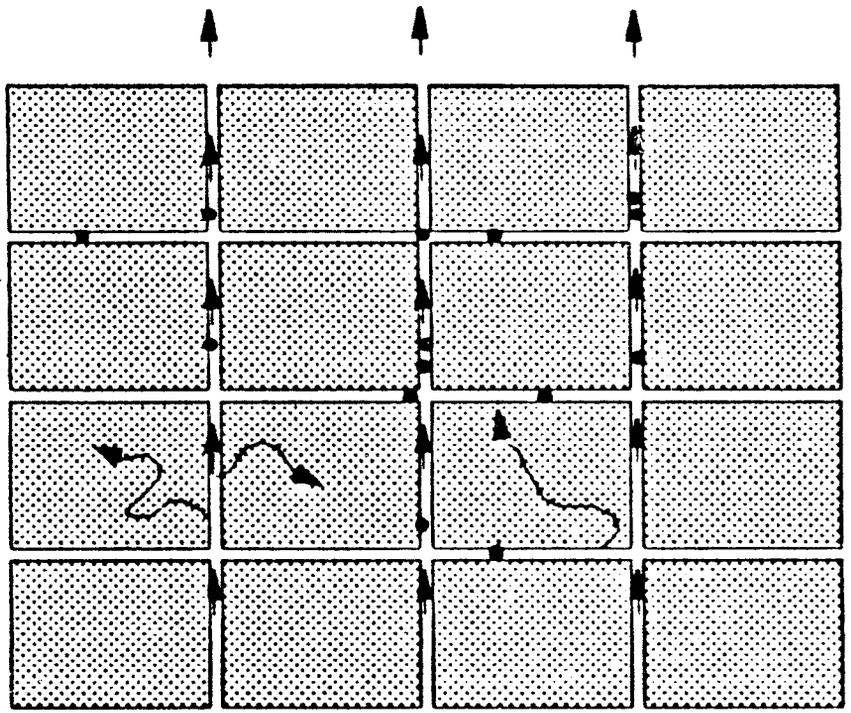


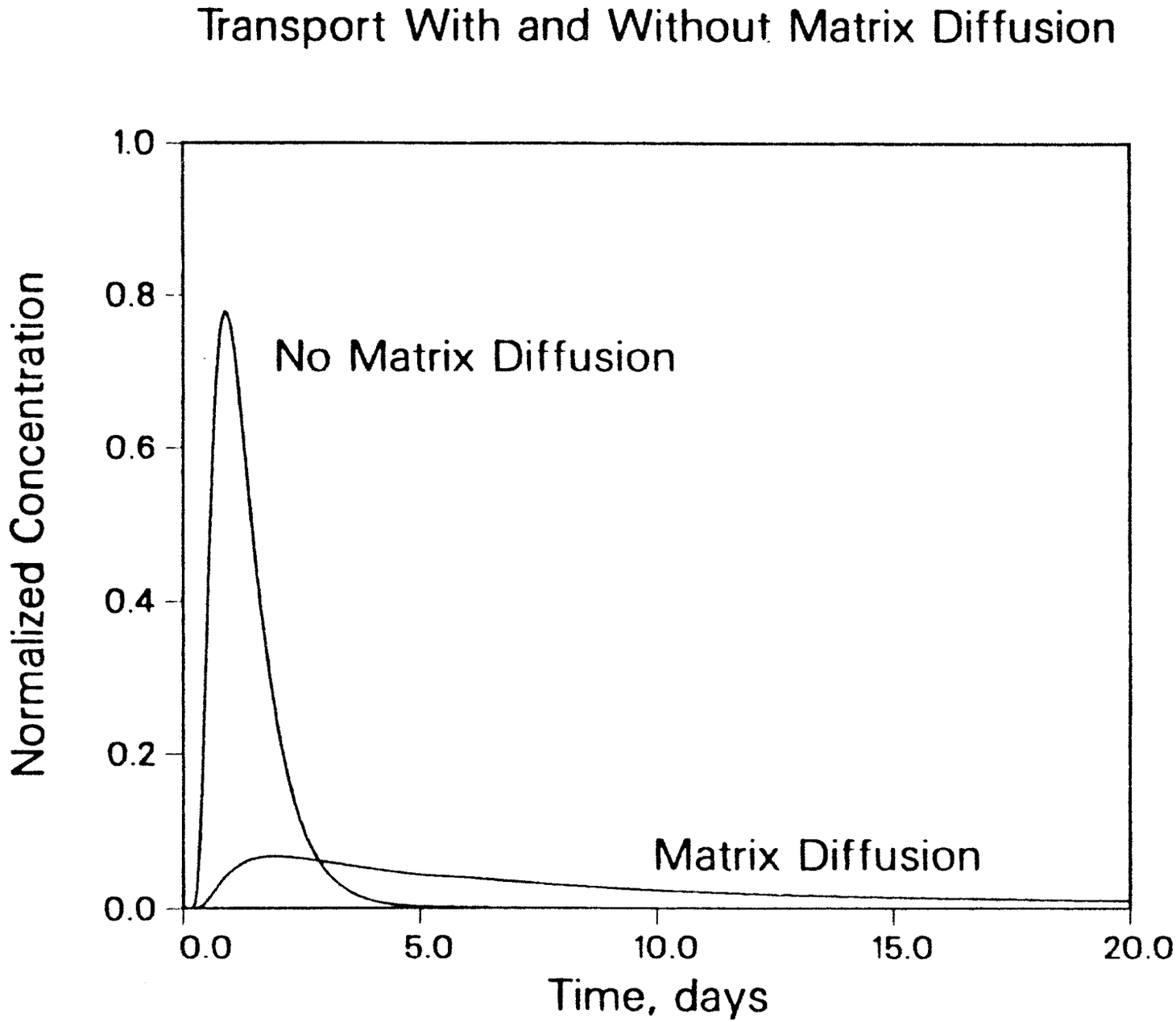

Breakthrough Curves - Different Flow Rates
Dimensionless Concentration
1.0
0.8
0.6
0.4
0.2
0.0
50 gpm
25 gpm
10 gpm
0.0
50.0
100.0
150.0
200.0
250.0
300.0
Produced Volume, cubic meters


LI Obs.
LI Pred.
Best Est. Obs.
Best Est. Pred.
Log [S(S$_{max}$ − S)$^{-1}$]
Log Conc. (μg mL^{-1})
2
1
0
−1
−2
−3
−1
0
1
2
3
4


Sorbing Tracer Behavior
Normalized Concentration
1.50
1.25
1.00
0.75
0.50
0.25
0.00
Conservative
R=1, R'=5
R=5, R'=5
0.0
1.0
2.0
3.0
4.0
5.0
Time, days

Colloid Transport Through Fractures




Transport With and Without Matrix Diffusion
Normalized Concentration
1.0
0.8
0.6
0.4
0.2
0.0
No Matrix Diffusion
Matrix Diffusion
0.0
5.0
10.0
15.0
20.0
Time, days

Issues Addressed by a Fourth Well

- QA Level 1 Core
- Density of Vertical Fractures

Other Issues:

- Fracture Mineral Coatings
- Scale Dependent Transport

Conclusions

- Test plan provides a means for validating matrix diffusion model
- LiBr tracer will justify the use of laboratory-determined sorption parameters to site-scale modeling
- Colloid mobility will be examined

D. Hoxie
USGS
Attachment 9

NET MOISTURE FLUX IN THE UNSATURATED ZONE AT YUCCA MOUNTAIN

--- UP, DOWN, OR SIDEWAYS?

D.T. Hoxie
U.S. Geological Survey
Denver, CO

DEFINITION

VOLUMETRIC FLUX

= Volume of fluid (liquid or gas) flowing through unit area in unit time

CENTERIOR ENERGY

Donald C. Shelton
Vice President - Nuclear
Davis-Besse

300 Madison Avenue
Toledo, OH 43652-0001
(419) 249-2300

Docket Number 50-346

License Number NPF-3

Serial Number 1899

March 21, 1991

United States Nuclear Regulatory Commission
Document Control Desk
Washington, D. C. 20555

Subject: Fire Protection - Essential Inverter Replacement Schedule Change

Gentlemen:

In Toledo Edison (TE) letter Serial Number 1757 dated February 16, 1990, TE committed to install four new essential 125 VDC inverters based on a 10 CFR Part 50, Appendix R, Fire Protection analysis which concluded that the existing four essential inverters (YV1, YV2, YV3, YV4) were not coordinated (i.e., a single fault could result in the inverter being lost due to input fuse failure). The coordination concern was eliminated during the sixth refueling outage by installing two new inverters, modifying the remaining two inverters and instituting additional operator actions. The remaining two new inverters were scheduled to be installed during the seventh refueling outage.

TE's intent to install the remaining two inverters during the seventh refueling outage is also reflected in Nuclear Regulatory Commission (NRC) Inspection Report 89-19, dated September 29, 1989, documenting closure of Licensee Event Report 84-010, and in Inspection Report 90-002, dated March 30, 1990, documenting closure of Licensee Event Report 87-013.

TE has determined that installation of the remaining two inverters during the seventh refueling outage is no longer necessary. In addition to the changes and modifications performed during the sixth refueling outage, several plant improvements have been implemented since 1985 which diminish the need for such replacement. These improvements include modifications to the integrated control system and non-nuclear instrumentation system to prevent a plant trip on loss of the old inverters, addition of manual transfer capabilities to an alternate power source for the old inverters, and improved plant maintenance practices which should limit the occurrence of power losses to the old inverters.

9103290189 910321
PDR ADOCK 05000346
P PDR

A006

As indicated above, the present configuration (two old and two new inverters), combined with manual operator actions described in the Fire Hazards Analysis Report (Revision 12), fully complies with the applicable requirements of 10 CFR 50, Appendix R. Therefore, TE does not intend to install the remaining two new inverters.

If you have any questions, please contact Mr. R. W. Schrauder, Manager Nuclear Licensing, at (419) 249-2366.

Very truly yours,

KBR/mmb

cc: P. M. Byron, DB-1 NRC Senior Resident Inspector
A. B. Davis, Regional Administrator, NRC Region III
J. R. Hall, NRC Senior Project Manager
M. D. Lynch, NRC Senior Project Manager
Utility Radiological Safety Board

March 22, 1991
RBG - 34,651
File Nos. G9.5, G9.25.1.4

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, D. C. 20555

Gentlemen:

River Bend Station - Unit 1
Docket No. 50-458

Enclosed is Gulf States Utilities Company's Special Report concerning a valid failure of the Division I diesel generator at River Bend Station. This report is submitted pursuant to River Bend Technical Specifications 4.8.1.1.3 and 6.9.2.

Sincerely,

W. H. Odell
Manager-Oversight
River Bend Nuclear Group

LAE/PDG/GAB/DCH/TES

cc: U.S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 1000
Arlington, TX 76011

NRC Resident Inspector
P.O. Box 1051
St. Francisville, LA 70775

INPO Records Center
1100 Circle 75 Parkway
Atlanta, GA 30339-3064

Mr. C. R. Oberg
Public Utility Commission of Texas
7800 Shoal Creek Blvd., Suite 400 North
Austin, TX 78757

9103290199 910322
PDR ADOCK 05000458
S PDR

IE22

SPECIAL REPORT

At 0030 on 02/20/91 during the performance of scheduled monthly surveillance test, STP-309-0201, "Division I Diesel Generator Operability Test", the jacket water and lube oil temperatures were observed increasing above their normal ranges. The diesel had been running at 100% load for 30 minutes when this occurred. To prevent the overheating of diesel components, Operations reduced the load, opened the output breaker and secured the diesel generator. During this surveillance test, the load was not maintained for greater than 60 minutes. Therefore, in accordance with Regulatory Guide 1.108 position C.2.e.6, this is considered a Valid Failure. Consequently, this Special Report is submitted pursuant to Regulatory Guide 1.108, position C.3.b and Technical Specification 4.8.1.1.3.

INVESTIGATION

The planned February Division I diesel generator outage began at 0750 on 01/19/91 when the diesel was placed in the maintenance mode. After completion of the scheduled preventative maintenance tasks, the diesel was restored to operational mode at 1640. The diesel was started at 2334 and loaded to 100% (greater than 3000 kw) at 2350 per STP-309-0201. Normal observation of instrumentation indicated that cooling of the jacket water system was inadequate. At this time the Operator in the diesel building manually adjusted the temperature control valve, 1EGT*TCV20A, in the "close bypass" direction. No temperature changes were observed following this adjustment. A decision was made to reduce the load and secure the diesel to eliminate any possible damage or a diesel trip due to high lube oil or jacket water temperatures.

After the diesel had cooled, a test run was performed at 0400 on 02/20/91. During this run a careful examination of lube oil and jacket water temperatures was performed. When the DG was loaded to 100%, temperatures again started rising above the normal ranges (jacket water 'JW' = 183 degrees F and lube oil 'LO' = 180 degrees F) and manual repositioning of the temperature control valve (TCV) was again performed. This time the valve repositioned itself and temperatures immediately decreased (JW = 133 degrees F, LO = 167 degrees F). The TCV was then repositioned back to its normal setting, and the diesel was run for over 60 minutes at full load with proper temperatures being maintained (JW = 157 to 162 degrees F and LO = 173 to 175 degrees F).

The investigation into the cause of the temperature increase indicated that the thermostatic temperature control valve appeared not to open to allow jacket water to flow through heat exchanger, 1EGT*E1A.

CAUSE OF FAILURE

An inspection of the valve internals was performed to determine the nature of the malfunction. The poppet cylinder and seat insert were in good condition. There was adequate clearance around the poppet cylinder, due to the deletion of an o-ring (reference NRC Information Notice 82-56), thus preventing binding at this location.

In this valve the return spring (or load spring) seats the valve body on the underside of the poppet. Its function is to close the valve until temperature rises to the level where thermostatic elements, or power pills, open the valve. When the valve opens, the stem, which houses the power pills, slides inside the load spring. There was evidence of some resistance to sliding between the retainer and stem. Surface discoloration, scratching, and small nicks were found on the stem. This is indicative of binding. Disassembly of the stem found debris (fine dirt and small flakes of rust) inside. In removing 2 of the 3 power pills, from the spacer sleeve in the stem, resistance to movement was also found. These two power pills were unable to properly function.The root cause of the failure was debris in the temperature control valve internals. All interior surfaces of the valve were cleaned. All parts internal to the stem, where binding had occurred, were replaced. The new power pills were benched tested in a bath of water to ensure they began to open at 160 degrees F and were fully open at 175 degrees F.

Note that a different rust inhibitor has been used in the jacket water systems since refueling outage 2, 06/01/89. During refueling outage 3, 10/90 - 11/90, the jacket water was drained and refilled several times. A "feed and bleed" was later performed which decreased the particulate iron oxide from 220 PPM to 12.5 PPM. Monitoring of jacket water samples has indicated the level of iron oxide particles has remained at the level of 12.5 PPM since refueling outage 3.

The valve was reassembled and a test run of the diesel was performed at 1644 on 02/21/91. During this run, it was necessary to manually adjust the TCV. However, this is normally expected after the valve has been overhauled.

Performance of STP-309-0201, to demonstrate diesel operability, began with a start at 2103 and a successful completion at 2239 on 02/21/91. During this run, the TCV operated satisfactorily and maintained jacket water and lube oil temperatures within their normal ranges, 157 to 162 degrees F and 173 to 175 degrees F, respectively.

CORRECTIVE ACTION

Cleaning and replacement of internal parts on the temperature control valve satisfactorily restored it to a reliable condition. A preventive maintenance task (ME03585) was initiated to have the Division I temperature control valve, 1EGT*TCV20A, disassembled and inspected each refueling outage to eliminate future build up of debris resulting in valve movement restriction. A similar preventive maintenance task (ME03586) was initiated to perform the same work on the Division 2 temperature control valve, 1EGT*TCV20B.

Since this was the 2nd failure of the Division I diesel generator within the last 20 valid tests, the test frequency was changed from once in 31 days to once in 7 days per Tech. Spec. Table 4.8.8.1.2-1. STP-309-0201 was subsequently performed with successful starts at 0134 and 1411 on 02/22/91, and 1328 on 02/27/91. After these tests were satisfactorily completed, the number of valid failures in the last 20 valid tests was reduced to one. This allowed the frequency of surveillance test, STP-309-0201, to be changed back to once in 31 days.

OPERATING START/TEST DATA

DIESEL GENERATOR MARK NUMBER:

1EGS*EG1A

LENGTH OF TIME DIESEL GENERATOR
WAS OUT OF SERVICE:

46 HOURS

SURVEILLANCE INTERVAL FOLLOWING FAILURE:

DIVISION I:	MONTHLY
DIVISION II:	MONTHLY
DIVISION III:	MONTHLY

TEST INTERVALS CONFORM TO TECHNICAL SPECIFICATIONS? YES

FAILURES FOR DIVISION I:

1 VALID FAILURE IN THE LAST 20 VALID TESTS

2 VALID FAILURES IN THE LAST 100 VALID TESTS

FAILURES FOR DIVISION II:

0 VALID FAILURES IN THE LAST 20 VALID TESTS

2 VALID FAILURES IN THE LAST 100 VALID TESTS

FAILURES FOR DIVISION III:

1 VALID FAILURE IN THE LAST 20 VALID TESTS

4 VALID FAILURES IN THE LAST 100 VALID TESTS

CUMULATIVE FAILURES FOR ALL RIVER BEND DIESEL GENERATORS:

4 VALID FAILURES IN THE LAST 100 VALID TESTS

March 21, 1991

Docket No. 50-333

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Mr. John C. Brons
Executive Vice President - Nuclear Generation
Power Authority of the State of New York
123 Main Street
White Plains, New York 10601

Dear Mr. Brons:

SUBJECT: PERSONNEL ASSIGNMENT

This is to inform you that effective April 1, 1991, Mr. Brian C. McCabe will be assigned Project Manager of the James A. FitzPatrick Nuclear Power Plant. Mr. McCabe can be reached on 492-1423.

Sincerely,

ORIGINAL SIGNED BY.

Robert A. Capra, Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

cc: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D C. 20666

March 21, 1991

Docket No. 50-333

Mr. John C. Brons
Executive Vice President - Nuclear Generation
Power Authority of the State of New York
123 Main Street
White Plains, New York 10601

Dear Mr. Brons:

SUBJECT: PERSONNEL ASSIGNMENT

This is to inform you that effective April 1, 1991, Mr. Brian C. McCabe will be assigned Project Manager of the James A. FitzPatrick Nuclear Power Plant. Mr. McCabe can be reached on 492-1423.

Sincerely,

Robert A. Capra

Robert A. Capra, Director
Project Directorate I-1
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Docket Nos. 50-313, 50-368
and 50-382, 50-416

MAR 2 1 1991

DESIGNATED ORIGINAL

Mr. William Cavanaugh, President
and Chief Executive Officer
Entergy Operations, Inc.
Echelon One
1340 Echelon Parkway
Jackson, Mississippi 39213

Dear Mr. Cavanaugh:

SUBJECT: OPERATIONAL EVENTS WHILE SHUTDOWN

The NRC has just issued Information Notice 91-22, "Four Plant Outage Events Involving Loss of AC Power or Coolant Spills," which addresses recent events that occurred during shutdown operations.

The chief purpose of this information notice is to notify each licensee that the high rate of precursor events to loss of decay heat removal during shutdown is a source of concern to the NRC. All of the events discussed in this information notice occurred during a one-week period in March 1991. Because of the potential for loss of a critical safety function in these and similar events, I believe a high level of management attention is required in the planning, coordination, and execution of shutdown operations.

While this information notice does not require specific licensee action or response, I urge you to give this important matter your personal attention.

Sincerely,

Original signed by
Thomas E. Murley

Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

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MAR 2 1 1991

Docket Nos. 50-313, 50-368
and 50-382, 50-416

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and Chief Executive Officer
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Echelon One
1340 Echelon Parkway
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Sincerely,

T Murley

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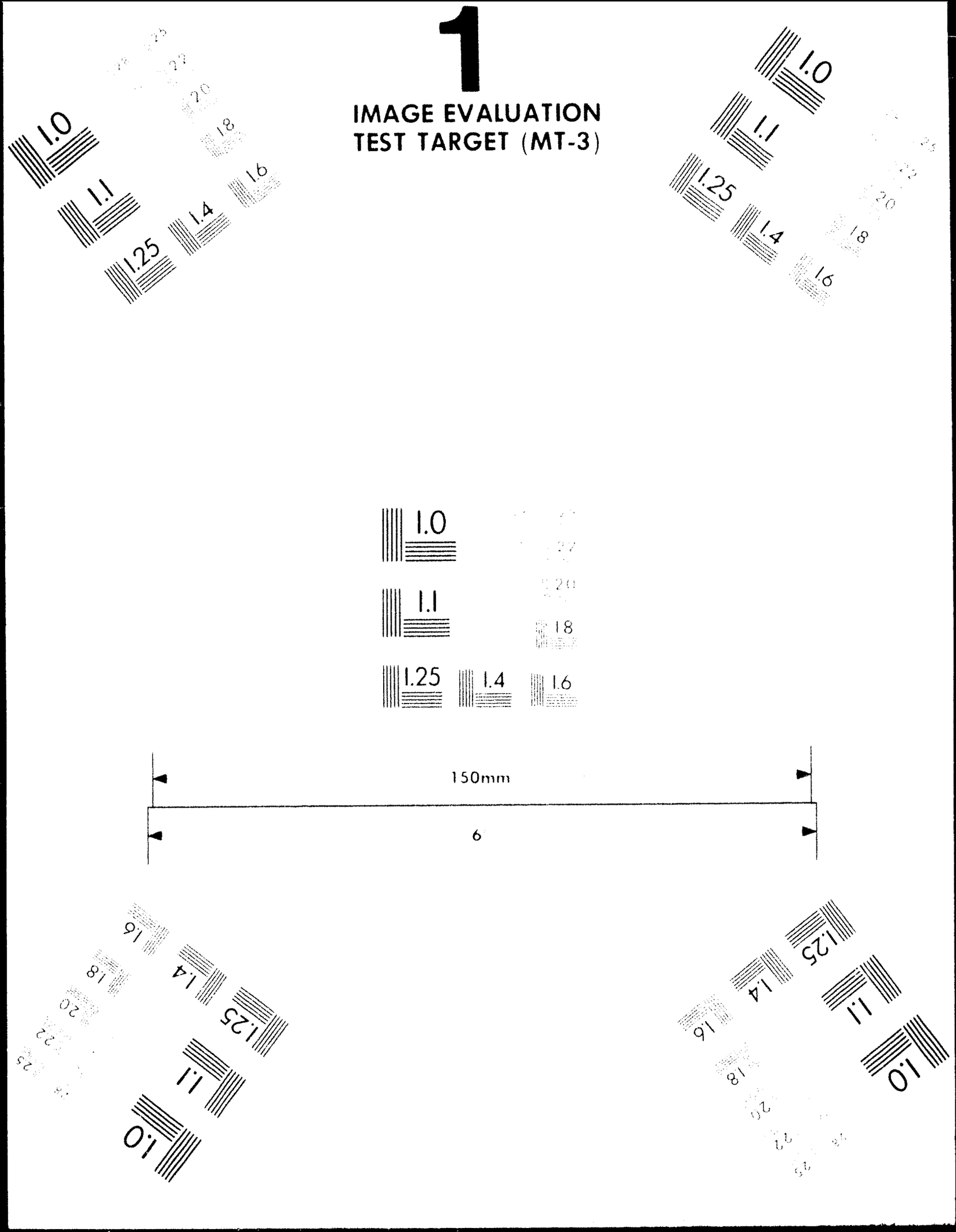
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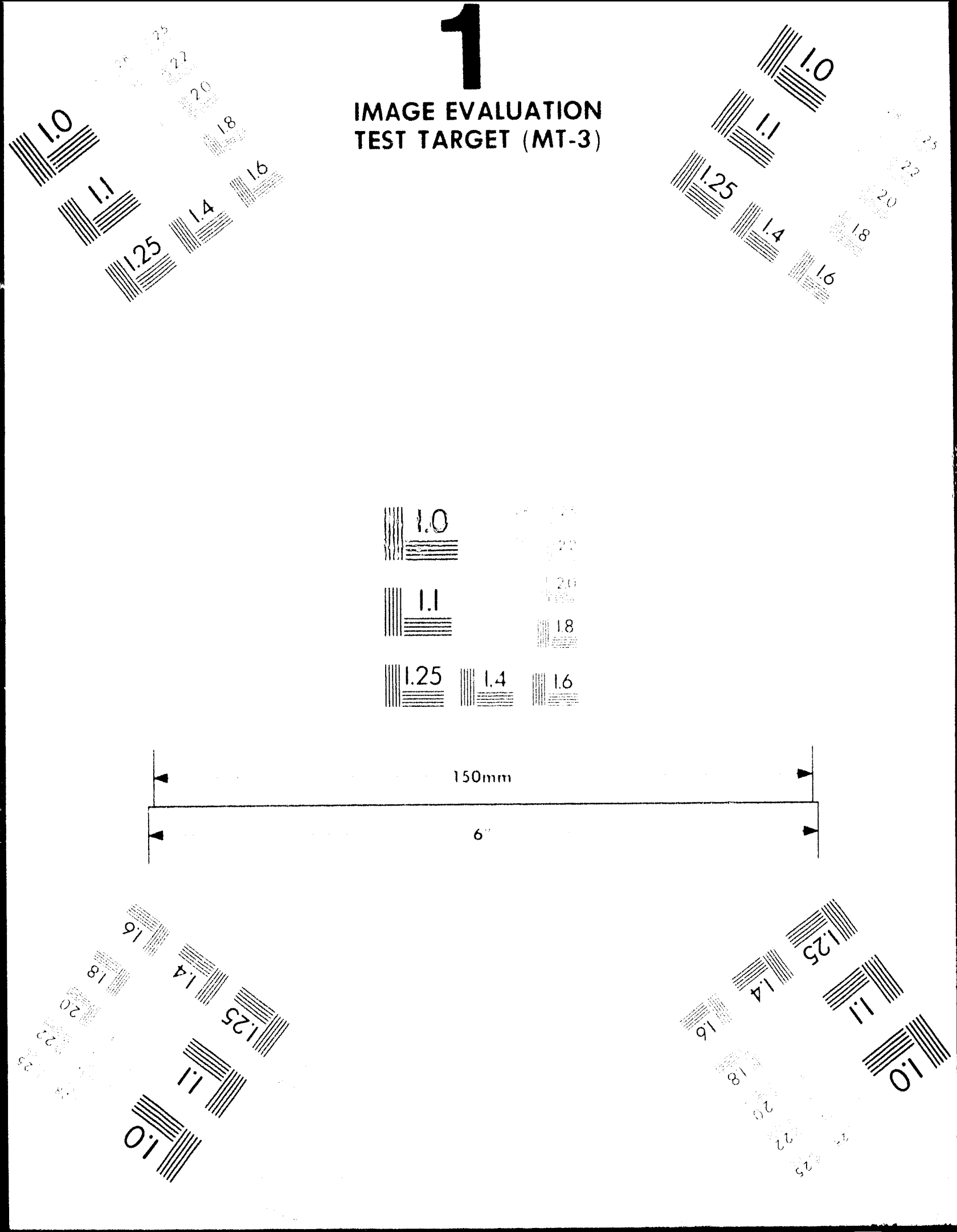
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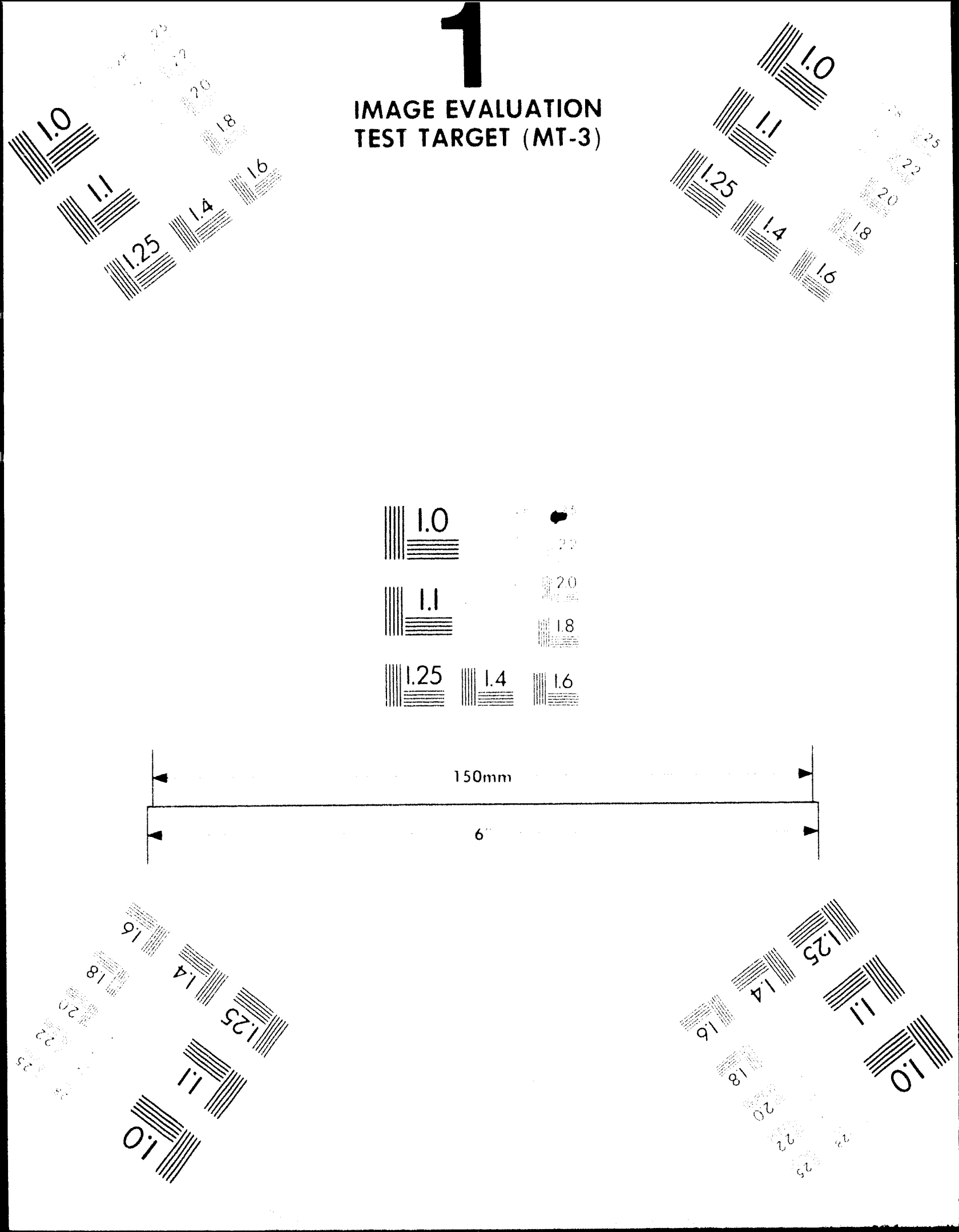
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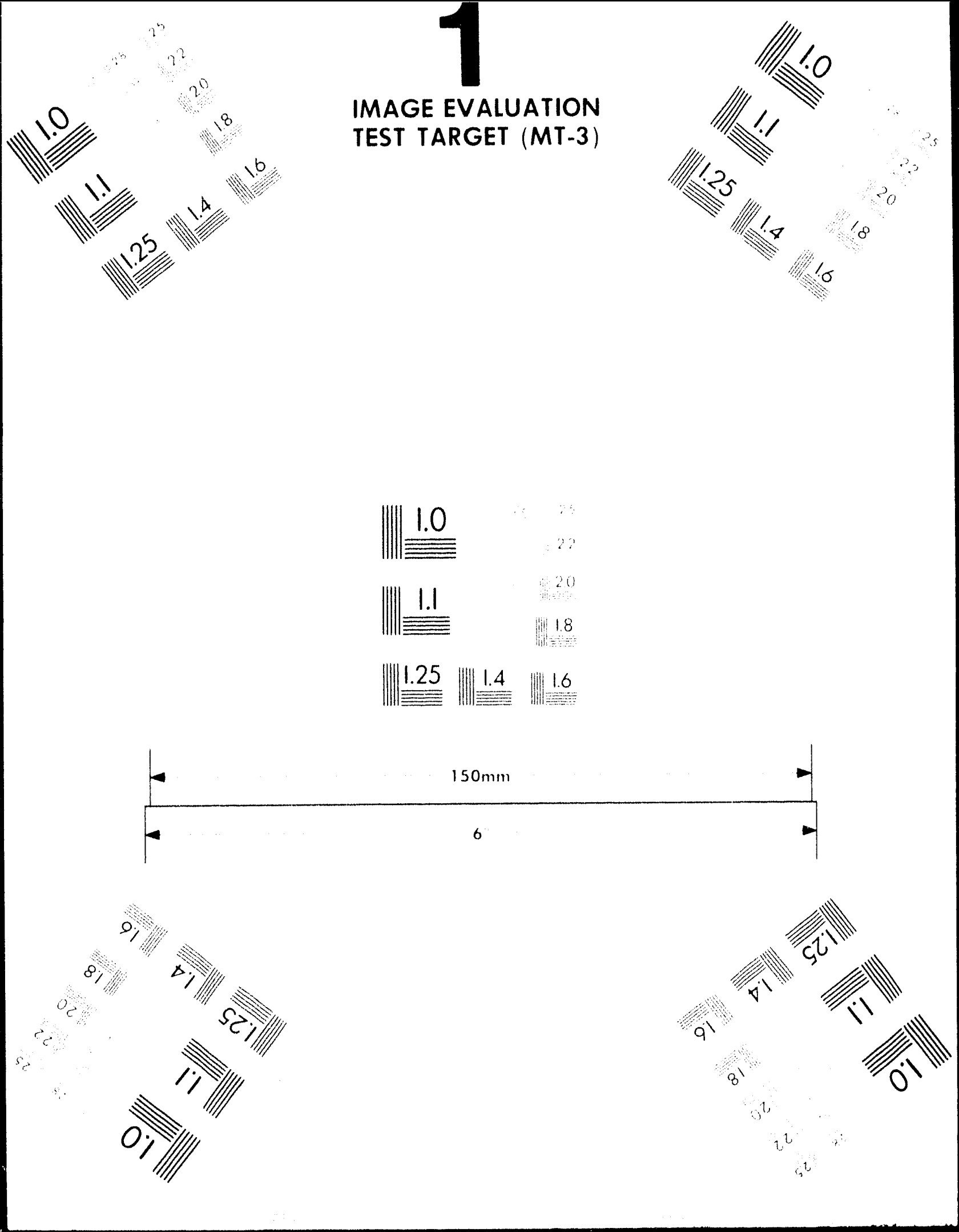
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