UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-8064

September 4, 2002

Mr. C. L. Terry, Senior Vice President
& Principal Nuclear Officer
TXU Generation Management Company LCC,
Managing General Partner for TXU Generation
Company LP
ATTN: Regulatory Affairs Department
P.O. Box 1002
Glen Rose, Texas 76043

SUBJECT: NRC TRIENNIAL FIRE PROTECTION INSPECTION REPORT 50-445/02-03; 50-446/02-03

Dear Mr. Terry:

On February 8, 2002, the NRC completed an inspection at your Comanche Peak Steam Electric Station, Units 1 and 2. The enclosed report documents the inspection findings, which were discussed with Mr. James Kelley, Vice President, Nuclear Engineering and Support, and other members of your staff in a telephone exit meeting conducted on September 3, 2002.

This inspection examined activities conducted under your license as they relate to implementation of your NRC-approved fire protection program. Within these areas, the inspection consisted of selected examination of procedures and representative records, observations of activities, and interviews with personnel.

Based on the results of this inspection, the NRC identified one finding, which remains unresolved pending the determination of its safety significance using the Significance Determination Process described in NRC Inspection Manual Chapter 0609. The issue has no immediate safety impact, as the licensee took immediate and effective action to correct the problem.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response will be made available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Charles S. Marschall, Chief
Engineering and Maintenance Branch
Division of Reactor Safety

Dockets: 50-445; 50-446
Licenses: NPF-87; NPF-89

Enclosure:
NRC Inspection Report
50-445/02-03; 50-446/02-03

cc w/enclosure(s):
Roger D. Walker
Regulatory Affairs Manager
TXU Generation Company LP
P.O. Box 1002
Glen Rose, Texas 76043

George L. Edgar, Esq.
Morgan, Lewis & Bockius
1800 M. Street, NW
Washington, D.C. 20036

G. R. Bynog, Program Manager/
Chief Inspector
Texas Department of Licensing & Regulation
Boiler Division
P.O. Box 12157, Capitol Station
Austin, Texas 78711

County Judge
P.O. Box 851
Glen Rose, Texas 76043

Chief, Bureau of Radiation Control
Texas Department of Health
1100 West 49th Street
Austin, Texas 78756-3189

John L. Howard, Director
Environmental and Natural Resources Policy
Office of the Governor
P.O. Box 12428
Austin, Texas 78711-3189

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Regional Administrator **(EWM)**
DRP Director **(KEB)**
DRS Director **(EEC)**
Senior Resident Inspector **(DBA)**
DRP Branch Chief, DRP/A **(WDJ)**
Senior Project Engineer, DRP/A **(CJP)**
DRP Chief, Technical Support Staff, DRP/TSS **(PHH)**
RITS Coordinator **(NBH)**
Scott Morris **(SAM1)**
CP Site Secretary **(LCA)**

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Dockets:	50-445; 50-446
Licenses:	NPF-87; NPF-89
Report No.:	50-445/02-03; 50-446/02-03
Licensee:	TXU Electric
Facility:	Comanche Peak Steam Electric Station, Units 1 and 2
Location:	FM-56 Glen Rose, Texas
Dates:	February 4 - 8, 2002
Team Leader	R. L. Nease, Senior Reactor Inspector Engineering and Maintenance Branch
Inspectors:	C. E. Johnson, Senior Reactor Inspector Engineering and Maintenance Branch R. P. Mullikin, Senior Reactor Inspector Engineering and Maintenance Branch N. F. O'Keefe, Senior Resident Inspector Project Branch A J. D. Hanna, Resident Inspector Project Branch B
Accompanying Personnel:	J. L. Taylor, Reactor Inspector Engineering and Maintenance Branch R. E. Deem, Contractor Brookhaven National Laboratories
Approved By:	Charles S. Marschall, Chief Engineering and Maintenance Branch Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000445-02-03, IR 05000346-02-03; TXU Electric; on 02/04/2002-02/08/2002; Comanche Peak Steam Electric Station; Units 1 and 2. Triennial Fire Protection Inspection.

The inspection was conducted by a team of three regional inspectors, one senior resident inspector, one resident inspector, one contractor, and one accompanying NRC Region IV personnel. The inspection identified one finding, which was a violation of NRC regulatory requirements. The significance of this finding has yet to be determined; therefore, the finding remains unresolved. The significance of issues is indicated by their color (green, white, yellow, red) and will be determined using the Significance Determination Process described in NRC Inspection Manual Chapter 0609. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG 1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. Inspector-Identified Findings

Cornerstone: Mitigating Systems

TBD. A violation of Comanche Peak Steam Electric Station Technical Specification, Section 5.4.1, was identified for failure to establish and maintain adequate procedures implementing the fire protection program. In particular, Abnormal Procedure ABN-803A, "Response to a Fire in the Control Room or Cable Spreading Room," did not direct operators to transfer control of the Train B power operated relief valve from the control room, leaving it vulnerable to spurious operation in the event of a fire in the control room envelope requiring control room evacuation and remote shutdown (Section 1R05.4).

The significance of this finding has yet to be determined; therefore, the finding remains unresolved.

Report Details

1. REACTOR SAFETY

1R05 Fire Protection

The purpose of this inspection was to review the Comanche Peak Steam Electric Station fire protection program for selected risk significant fire areas. Emphasis was placed on verification of the licensee's post-fire safe shutdown capability. The inspection was performed in accordance with the new Nuclear Regulatory Commission (NRC) reactor oversight process using a risk-informed approach for selecting the fire areas and attributes to be inspected. The team used the licensee's "Individual Plant Examination of External Events for Severe Accident Vulnerabilities, Comanche Peak Steam Electric Station," dated June 1995, to choose several risk-significant areas for detailed inspection and review. The fire areas chosen for review during this inspection were:

- SB - Unit 1 Safeguards Building Elevations 773' through 841'-6"

- SD - Unit 1 safeguards building electrical equipment room, Train A switchgear on the 810'-6" elevation

- SE - Unit 1 Safeguards Building Elevations 831'-6" and 852'-6"

- AAS - Elevations 778' through 886'-6" of the auxiliary building and some areas within the electrical and control building, and the fuel building

For each of the selected fire areas, the team focused the inspection on the fire protection features and on the systems and equipment necessary for the licensee to achieve and maintain safe shutdown conditions in the event of a fire in those fire areas.

.1 Systems Required to Achieve and Maintain Post-Fire Safe Shutdown

a. Inspection Scope

The team reviewed the licensee's piping and instrumentation diagrams, safe shutdown equipment list, safe shutdown design basis document, and the post-fire safe shutdown analysis to verify whether the licensee's shutdown methodology had properly identified the components and systems necessary to achieve and maintain safe shutdown conditions for equipment in the fire areas selected for review. The team also reviewed the licensee's procedures for achieving and maintaining safe shutdown in the event of a fire to verify that the safe shutdown analysis provisions were properly implemented. The team focused on the following functions that must be ensured to achieve and maintain post-fire safe shutdown conditions: (1) reactivity control capable of achieving and maintaining cold shutdown reactivity conditions, (2) reactor coolant makeup capable of maintaining the reactor coolant level within the level indication in the pressurizer, (3) reactor heat removal capable of achieving and maintaining decay heat removal, (4) supporting systems capable of providing all other services necessary to permit extended operation of equipment necessary to achieve and maintain hot shutdown conditions, and (5) process monitoring capable of providing direct readings necessary to control the above functions.

b. Findings

No findings of significance were identified.

.2 Fire Protection of Safe Shutdown Capability

a. Inspection Scope

The team verified that at least one post-fire safe shutdown success path was free of fire damage in the event of a fire in the selected fire areas. Specifically, the team examined the separation of safe shutdown cables, equipment, and components within the same fire areas, and reviewed the licensee's methodology for meeting the requirements of 10 CFR 50.48 and NRC Branch Technical Position 9.5-1. In addition, the team reviewed license documentation, such as NRC safety evaluation reports, the Comanche Peak Steam Electric Station Updated Final Safety Evaluation Report, submittals made to the NRC by the licensee in support of the NRC's review of their fire protection program, and deviations from NRC regulations to verify that the licensee met license commitments.

b. Findings

No findings of significance were identified.

.3 Post-Fire Safe Shutdown Circuit Analysis

a. Inspection Scope

On a sample basis, the team verified that cables and circuits for equipment required to achieve and maintain hot shutdown conditions in the event of a fire in selected fire areas had been properly identified and either adequately protected from the potentially adverse effects of fire damage or analyzed to show that fire-induced faults (e.g., hot shorts, open circuits, and shorts to ground) would not prevent safe shutdown. During the inspection, a sample of redundant components associated with systems required to achieve and maintain hot shutdown conditions were selected for review. The sample included components associated with the auxiliary feedwater, safety injection, reactor coolant system makeup, component cooling water, service water cooling, and pressurizer power-operated relief valves. From this list of components, the team reviewed data depicting the routing of power and control cables associated with each of the selected components. Additionally, the team verified, on a sample basis, that circuit breaker coordination and fuse protection were acceptable as a means of protecting the power sources of the designated safe shutdown equipment.

b. Findings

No findings of significance were identified.

.4 Alternative Safe Shutdown Capability and Implementation

a. Inspection Scope

The team reviewed the licensee's systems required to achieve alternative safe shutdown to determine if the licensee had properly identified the components and systems necessary to achieve and maintain safe shutdown conditions from stations other than the control room. The team also focused on the adequacy of the systems to perform reactor pressure control, reactor makeup, decay heat removal, process monitoring, and support system functions. The team reviewed Abnormal Procedure ABN-803, "Response to a Fire in the Control Room or Cable Spreading Room," Revision 5, used by operators to shutdown the reactor in the event a fire in the control room results in a control room evaluation. The team also stepped through the procedure with licensed operators to determine the adequacy of the procedure to direct safe shutdown activities from remote shutdown locations.

b. Findings

The team identified that Abnormal Procedure ABN-803A failed to direct operators to transfer control of the B train pressurizer power-operated relief valve (PORV) from the control room to the remote shutdown panel, leaving it vulnerable to spurious operation as a result of fire damage to control cables. A violation of Comanche Peak Steam Electric Station Technical Specification, Section 5.4.1, was identified for failure to establish and maintain adequate procedures covering activities associated with fire protection program implementation. The determination of the significance of this violation was referred to the NRC's Office of Nuclear Reactor Regulation; therefore, this issue remains unresolved pending the completion of that effort (50-445/0203-01).

The team noted that Analysis WCAP 11331, "Comanche Peak Steam and Electric Station Thermal/Hydraulic Analysis of Fire Safe Shutdown Scenarios," identified that a spuriously open pressurizer PORV was a limiting fault. This analysis calculated that the PORV must be shut within 3 minutes to maintain pressurizer level within the indicating band, a requirement of 10 CFR Part 50, Appendix R, Section III.L. Upon questioning, the licensee performed a calculation that revised this time to 5 minutes.

The team reviewed Abnormal Procedure ABN-803A, which would be used in the event that a fire in the control room envelope was of the magnitude to require evacuation of the control room and shutdown of the reactor from the remote shutdown panel. The team found that in performing an alternative shutdown from outside the control room, Abnormal Procedure ABN-803A directed actions to establish control of the Train A PORV, but not the Train B PORV. Specifically, the Train A PORV transfer switch was procedurally required to be placed in the hot shutdown panel position, which replaced the portion of the normal control circuit physically located in the control room with an alternative shutdown control circuit, which was protected from fire damage. However, a similar action was not procedurally directed for the Train B PORV, even though this capability was installed in the plant. The licensee indicated that power was expected to remain available to operate the Train B PORV during this scenario, and that spurious opening of the valve was possible as a result of a fire in the control room. Section 5.4.1

of Technical Specifications for Comanche Peak Steam Electric Station, Units 1 and 2, requires that written procedures covering fire protection program implementation be established, implemented, and maintained. The licensee failed to properly implement the requirement to transfer control of the Train B PORV from the control room to the remote shutdown panel (in the event of a fire requiring control room evacuation), as described in their fire protection program. This was a violation of Technical Specification, Section 5.4.1. The licensee entered this issue into their corrective action program as Smart Form SMF-2002-000354-00, and immediately revised Abnormal Procedure ABN 803A to include instructions to transfer control of the Train B PORV.

This issue was determined to be more than minor, because it is associated with the reactor safety mitigating systems cornerstone and affects the cornerstone objective as described in NRC Manual Chapter 0612, Appendix B. Specifically, this finding affects the licensee's capability to mitigate the consequences of a fire in the control room in order to achieve and maintain safe shutdown. Abnormal Procedure ABN-803A would be used by operators in the event of a fire in the control room or in the cable spreading room. The team leader and Region IV senior reactor analyst performed a Phase 2 risk assessment postulating a fire in the control room only because it had a larger ignition frequency and no automatic suppression, thus, yielding more conservative results. The Phase 2 risk evaluation performed using the NRC's Significance Determination Process (SDP) described in Manual Chapter 0609, Appendix F, indicated that the significance of this finding could be greater than very low (GREEN). Due to uncertainties in the Phase 2 SDP fire protection modeling for this finding, a Phase 3 significance determination was deemed necessary. The Phase 3 significance determination of this violation was referred to the NRC's Office of Nuclear Reactor Regulation. This issue remains unresolved pending the completion of that effort (50-445/0203-01).

.5 Emergency Communications

a. Inspection Scope

The team reviewed the adequacy of the communication system to support plant personnel in the performance of alternative safe shutdown functions and fire brigade duties. The team verified that the required number of radios was available for use and maintained in working order. The inspectors performed a detailed review of the electrical power supplies to both the radio repeater and Gaitronics™ systems. This review was performed to ensure that a postulated fire could not disable multiple trains of electrical power to communications equipment.

b. Findings

No findings of significance were identified.

.6 Emergency Lighting

a. Inspection Scope

The team reviewed the emergency lighting system required for safe shutdown activities to verify it was adequate for supporting the performance of manual actions required to achieve and maintain hot shutdown conditions, and for illuminating access and egress routes to the areas where manual actions are required. The team reviewed repetitive tasks for testing and test data trending to verify that the individual battery operated units were capable of supplying sufficient illumination

b. Findings

No findings of significance were identified.

.7 Cold Shutdown Repairs

a. Inspection Scope

The team reviewed licensee procedures to determine whether repairs were required to achieve cold shutdown and to verify that the repair material was available onsite. The team verified that the licensee had pre-staged equipment necessary to perform the repairs in lockers, as required by procedure.

b. Findings

No findings of significance were identified.

.8 Fire Protection Systems, Features, and Equipment

a. For the selected fire areas, the team evaluated the adequacy of fire protection features, such as fire suppression and detection systems, fire area barriers, penetration seals, and fire doors. To do this, the team observed the material condition and configuration of the installed fire detection and suppression systems, fire barriers, and construction details and supporting fire tests for the installed fire barriers. In addition, the team reviewed license documentation, such as NRC safety evaluation reports, and deviations from NRC regulations and the National Fire Protection Association (NFPA) code to verify that fire protection features met license commitments.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

4OA2 Identification and Resolution of Problems

a. Inspection Scope

The team reviewed a sample of Smart Forms to verify that the licensee was identifying fire protection-related issues at an appropriate threshold and entering those issues into the corrective action program. A listing of Smart Forms reviewed is provided in the attachment to this report.

b. Findings

No findings of significance were identified.

4OA6 Meetings, including Exit

On, February 8, 2001, at the conclusion of the team's onsite inspection, the team leader debriefed Mr. James J. Kelley, and other licensee staff members on the preliminary inspection results.

On September 3, 2002, the team leader conducted a telephone exit meeting with Mr. James Kelley, Vice President, Nuclear Engineering and Support, and other licensee staff members, during which the results of this inspection were characterized.

The licensee was asked whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

ENCLOSURE

KEY POINTS OF CONTACT

Licensee

K. Apple, Fire Protection Equipment Maintenance Supervisor
H. Beck, Contractor
G. Beckett, Engineering Programs Consulting Engineer
O. Bhatty, Engineering Programs Senior Engineer
J. Boatwright, Reactor Safety Analyst
T. Daskam, Operations Procedure Writer
R. Dible, Contractor
C. Cotton, Operations Dayshift Manager
C. Gibson, Contractor
S. Karpyak, Risk and Reliability Engineering Supervisor
S. Lakdawala, Engineering Programs Manager
G. Merka, Senior Nuclear Specialist
J. Seawright, Regulatory Affairs Consulting Engineer
J. Stone, Operations Procedure Writer
D. Tirsun, Reactor Engineering Consulting Engineer

NRC

D. Loveless, Senior Reactor Analyst, Region IV
P. Qualls, Office of Nuclear Reactor Regulation
M. Salley, Office of Nuclear Reactor Regulation
S. Wong, Office of Nuclear Reactor Regulation

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-445/0203-01	URI	Failure to establish and maintain adequate procedures covering activities associated with fire protection program implementation (Section 1R05.4).

LIST OF DOCUMENTS REVIEWED

The following documents were selected and reviewed by the team to accomplish the objectives and scope of the inspection.

CABLE ROUTING DATA

Component	Component	Component	Component
1-8000A	1-8000B	1-8105	1-8106
1-8351A	1-8351B	1-8351C	1-8351D
1-8811A	1-8811B	1HV2491A	1HV2492A
1HV2493B	1HV2494B	1HV4286	1HV4287
1HV4393	1HV4394	1HV4395	1HV4396
1HV4524	1HV4524	1HV4525	1HV4526
1HV4527	1PCV455A		

CALCULATIONS

Number	Title	Revision
EE-CA-0008-157	Correct Coordination Curve for Service Water Pump Motor	2
ME-CA-0000-1086	Safe Shutdown Analysis	1

DRAWINGS

Drawing Number	Title	Revision
E1-0001	Plant One Line Diagram - Units 1 and 2	CP-22
E1-0001, Sheet A	Plant One Line Diagram - Unit 1 and Common- Distribution Panels	CP-17
E1-0064, Sheet 11	Nitrogen Operated Valve 1-PCV-0455A - Pressurizer Power Relief Valve	CP-6
E1-0064, Sheet 12	Nitrogen Operated Valve 1-PCV-0456 - Pressurizer Power Relief Valve	CP-6
M1-0202	Main Steam Reheat and Steam Dump	CP-15
M1-0203	Steam Generator Feedwater System	CP-27
M1-0206	Auxiliary Feedwater System	CP-18
M1-0229	Component Cooling Water System	CP-19
M1-0230	Component Cooling Water System	CP-26
M1-0231	Component Cooling Water System	CP-24

Drawing Number	Title	Revision
M1-0233	Station Service Water System	CP-35
M1-0234	Station Service Water System	CP-23
M1-0250	Reactor Coolant System	CP-29
M1-0251	Reactor Coolant System	CP-28
M1-0253	Chemical and Volume Control System	CP-10
M1-0254	Chemical and Volume Control System	CP-21
M1-0255	Chemical and Volume Control System	CP-22
M1-0256	Chemical and Volume Control System	CP-12
M1-0257	Chemical and Volume Control System	CP-24
M1-0260	Residual Heat Removal System	CP-26
M1-0261	Safety Injection System	CP-20
M1-0262	Safety Injection System	CP-22
M1-0263	Safety Injection System	CP-15
M1-0304	Ventilation Control Room Air Conditioning	CP-32
M1-0311	Ventilation Safety Chilled Water System	CP-26
M1-0313	Ventilation Control BLDG. UPS Area A/C Systems	CP-1
M1-1901	Penetration Seals - Typical Details	CP-4
M1-1902, Sheet 10	Penetration Seal Typical Detail #2	CP-1
M1-1902, Sheet 11	Penetration Seal Typical Detail #2A	CP-1
M1-1921	Fire Hazards Analysis Unit 1- Containment and Safeguards Buildings Plans at El 808’-0" and 810’-6"	CP-37
M1-1923	Fire Hazards Analysis Unit 1- Containment and Safeguards Buildings Plans at El 852’-6" and 860’-0"	CP-3
M1-1928	Fire Hazards Analysis Unit 1- Auxiliary and Electrical Control Bldg - El 807’-0" and 810’-6"	CP-2
M1-1931	Fire Hazards Analysis Unit 1- Auxiliary Building - El 842’-0" and 873’-6"	CP-3

Drawing Number	Title	Revision
SG-810-083-8, Sheets 8 & 9	Penetration Seal Map Rm. 103, Safeguard - Unit 1	CP-1
SG-852-103-20, Sheet 20	Penetration Seal Map Rm. 83, Safeguard - Unit 1	CP-1

ENGINEERING REPORTS

Number	Title	Revision/Date
ER-ME-038	Evaluation of Fire-Rated Penetration Seal Detail	1
WCAP 11331	Comanche Peak Steam and Electric Station Thermal/Hydraulic Analysis of Fire Safe Shutdown Scenarios	October 31, 1986

FIRE IMPAIRMENTS

01-0537	02-0055
02-0001	02-0535

PROCEDURES

Number	Title	Revision
ABN-803A-R5-7	Response to a Fire in the Control Room or Cable Spreading Room	5
ABN-804A-R3-1	Response to Fire in the Safeguards Building	3
ABN-805A-R4-1	Response to Fire in the Auxiliary Building or the Fuel Building	4
ABN-901	Fire Protection System Malfunction Abnormal Conditions Procedures Manual	6
EOP-0.0A	Reactor Trip or Safety Injection	7
ODA-403	Locked Component Control Operations Department Administration Manual	4
OWI-103	Locked Component Listings and Deviation Control	13
OWI-203	Operations Department Management Periodic Reviews	9
STA-724	Fire Reporting and Response	2
STA-727	Fire Brigade Station Administration Manual	4

Number	Title	Revision
1CP-PT-06-06	Preoperational Test Procedure, Fire Protection-CSR Halon	0

SMART FORMS

SMF-1999-000078	SMF-2002-000354
SMF-1999-000541	SMF-2002-000361
SMF-1999-000849	SMF-2002-000371
SMF-2000-000014	SMF-2002-000372
SMF-2002-000277	SMF-2002-000373
SMF-2002-000325	SMF-2002-000383
SMF-2002-000346	SMF-2002-000395

MISCELLANEOUS DOCUMENTS

CPSES ABN-803 Simulator Training Scenario, dated September 28, 2000

Comanche Peak Steam Electric Station Fire Protection Report, Unit 1 and 2, dated September 15, 1998

DBD-ME-020, “Fire Safe Shutdown Analysis,” dated 06/26/01

Design Modification DMRC 87-1-231, “Unit 1 Cable Spreading Room Halon System Upgrade,” approved December 21, 1987

Individual Plant Examination of External Events for Severe Accident Vulnerabilites, Comanche Peak Steam Electric Station, dated June 1995

NFPA 12A, “Halon 1301 Fire Extinguishing Systems,” 1980

NFPA 72E, “Automatic Fire Detectors,” 1974

NFPA 72E Code Compliance Review, Automatic Fire Detectors, 1978 Edition, prepared by IMPELL Corporation, dated April 1987

NUREG-0797, Supplement No. 9, "Safety Evaluation Report related to the operation of Comanche Peak Steam Electric Station, Units 1 and 2," dated March 1985

NUREG-0797, Supplement No. 21, "Safety Evaluation Report related to the operation of Comanche Peak Steam Electric Station, Units 1 and 2," dated April 1989

Operations Duty Ring Issue Log for February 5, 2002
Penetration Seal Schedule, ECE-M1-1900, Report 10 - Master Listing, Page 685, dated August 15, 1991

Specification 2323-MS-38E, “Requirements for Design, Furnishing, Delivery, Testing and Certification of the Halon Agent 1301,” Revision 1

Shift Managers Relief Checklist dated February 5, 2002

Test Number EGT-TP-87A-37, “Fire Brigade and Operations Radio System Operability Test,” Revision 0 performed on January 9, 1989

Test Number PPT-TP-92C-15, “Unit 2 Radio Coverage Test,” Revision 0, performed on December 21, 1992