Portland General Electric Company

Bart D. Withers Vice President

July 2, 1981

Trojan Nuclear Plant
Docket 50-344
License NPF-1

Mr. Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Sir:

Attached please find supplemental information regarding NUREG-0696 "Emergency Response Facilities" which was discussed at the June 15, 1981 meeting between the NRC and PGE. The Attachment covers five areas: 1) TSC Building, 2) TSC Computer System, 3) EOF Habitability, 4) EOF Data System, and 5) SPDS.

As discussed at the June 15, 1981 meeting, design descriptions of the Emergency Response Facilities were submitted to the NRC on October 21, 1980 and April 15, 1981 as well as in the Trojan Radiological Emergency Plan which was submitted on December 31, 1980. Subsequent to the first submittal, procurement and construction of the TSC and SPDS were initiated and are in progress to comply with the NRC-required implementation schedule.

Pursuant to your request, five copies of WCAP-9725 (proprietary) reports entitled "Westinghouse Technical Support Complex" are attached with this submittal. These reports were originally transmitted to the NRC by Westinghouse (T. M. Anderson's letter to J. R. Miller dated June 13, 1980). In conformance with the requirement of 10 CFR 2.790, as amended, of the Commission's regulations, we are attaching with this submittal an application for withholding from public disclosure and an affidavit by Westinghouse.

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Portland General Electric Company

Resolution and approval by the NRC staff in the areas discussed in the attachment are required by August 1, 1981, in order for us to comply with the October 1, 1982 completion date specified in NUREG-0696.

Sincerely,

Bart D. Withers
Vice President
Nuclear

Attachment

c: Robert A. Clark, Chief
Operating Reactors Branch No. 3
Division of Licensing
U. S. Nuclear Regulatory Commission

Lynn Frank, Director
State of Oregon
Department of Energy

TROJAN NUCLEAR PLANT
SUPPLEMENTAL INFORMATION ON NUREG-0696
EMERGENCY RESPONSE FACILITIES (TSC, EOF and SPDS)

A detailed design description of the Technical Support Center (TSC), Emergency Operations Facilities (EOF) and Safety Parameter Display System (SPDS) for the Trojan Nuclear Plant was initially submitted to the NRC on October 21, 1980 in response to the NRC letter of September 5, 1980. Draft NUREGs-0696 and -0660, NUREGs-0578, -0585, and -0654 (Revision 0) were reviewed and considered in the design of the Trojan Emergency Response Facilities for the October 21, 1980 PGE submittal. Utilization of the EOF and TSC was described in the Trojan Radiological Emergency Plan, which was submitted on December 31, 1980.

Due to the required implementation schedule of January 1, 1982 at that time, procurement of the TSC computer system and the construction of the TSC building were commenced in May and December of 1980 respectively, which was before the final issuance of NUREG-0696.

NUREG-0696 was issued in final form in February 1981. Supplemental information identifying exceptions to the final NUREG-0696 criteria was transmitted by PGE to the NRC on April 15, 1981 with appropriate justifications.

Pursuant to the June 15, 1981 meeting between the NRC and PGE, the following additional information is provided.

I. TSC Building

The identification of office spaces on the first floor of the TSC is described in Figure 1. The lunch/conference room located on the northwest corner will be assigned as the office space for the NRC personnel during emergency conditions. The remaining three office spaces will be used during normal and emergency conditions by PGE personnel. During emergency conditions the office/conference room in the southeast corner can be used for special conferences and meetings. Since the TSC is planned to be used by the Plant personnel during normal operations, the work area next to the command center is divided by temporary partitions about five feet in height as office spaces. This area can also be used for offices during emergency conditions. The command center, which is approximately 985 sq ft, is located on the southwest corner of the building and is equipped with accordian-type curtains to isolate that space from the remaining office spaces. The detailed dimensions and layout of the command center is described in WCAP-9725 (June 1980), Figure 2.1. As shown in Figure 1, the TSC Building is also equipped with a decontamination room to prevent radioactive contamination of the building because of personnel access during emergency conditions. Decontamination equipment as well as anticontamination clothing will be located in the decontamination room.

In case the need arises to travel between the TSC and the control room during emergency conditions, the optimum travel route is out the emergency exit or the entrance door of the north side of the TSC, east through the yard to the Turbine Building doors (Figure 2). There are three doors on the west side of the Turbine Building at El. 45 ft. and two doors on the south side. A person traveling to the control room is expected to take one of the Turbine Building doors, walk through the Turbine Building El 45 ft. floor to the Control Building and take either an elevator or stairs to the El. 93 ft. level where the control room is located. In case this travel route is not available, there are also stairways internal to the Turbine Building which connect all three floors to the access door from the yard. Each floor of the Turbine Building is connected to the Control Building floors by doors. Therefore, the access through the Turbine Building to the Control Building is assured. Travel through the Turbine Building will also provide shielding from radioactivity releases which may be occurring. In any case, the travel between the TSC and the control room can be conducted without passing through the security gate.

II. TSC Computer System

A general description of the TSC Computer System was provided in the October 21, 1980 PGE submittal, which was supplemented by WCAP-9725 (June 1980) for detailed design philosophy and descriptions (five copies of WCAP-9725 are attached to this submittal for your review). As discussed in the October 21, 1980 submittal, the Westinghouse Computer System located in the TSC will be utilized to accomplish the functions of data collection, data manipulation, and data display for the TSC, the SPDS and the Bypassed and Inoperable Status Indication (BISI) System. A description of the SPDS and BISI System is provided in Section V of this report.

The parameters to be included in the TSC Data System are delineated in Table 1. The parameters in Table 1 constitute critical parameters considered necessary for recognition and mitigation of accident conditions. Other parameters, which are specified in Regulatory Guide 1.97 (Revision 2) but not included in the Trojan TSC Data System, will be made available on demand to the TSC and EOF by telephone communication. Containment sump water temperature and Containment spray flow,which are not currently measured, will be inferred by the Containment temperature and Containment spray pump discharge pressure.

Qualifications of the components which provide input to the TSC Data System are presently being reviewed as part of the Regulatory Guide 1.97 review. Information regarding equipment qualifications will not be available until the Regulatory Guide 1.97 review is completed in 1982.

III. EOF HABITABILITY

The Trojan EOF is located in the Trojan Visitors Information Center (VIC), approximately 700 meters WSW of the Containment. A floor plan of the EOF is shown in Figure 3. The VIC possesses several advantages as an EOF location:

A. Its closeness to the Plant allows for fast activation, and permits the use of Plant personnel as the initial EOF staff.

B. It can serve as an Assembly/Decontamination Area for personnel evacuated from the Plant. Any personnel can then quickly travel to the Plant if needed.

C. The EOF staff, including the Emergency Response Manager (PGE Vice-President, Nuclear), is located close to the Plant, providing more effective control of the emergency.

D. NRC, FEMA, and State and county representatives are located close to the Plant.

E. The VIC is a good location from which to control site security.

F. The VIC has been used as the EOF since 1975. Plant personnel, as well as State and county emergency response personnel, are familiar with the facility and its location.

G. Using the VIC as an EOF helps to utilize existing structures near the Plant most efficiently.

H. A low wind frequency (0.06%) toward the WSW provides good assurance of EOF habitability.

The EOF is a well engineered structure that is designed to withstand a 100-yr. flood.

An analysis has been performed on the ability of the EOF to withstand a 100-yr wind. At this wind level, some damage to the EOF Building may occur, but will likely be limited to broken windows. However, the command, communications, and dose assessment functions of the EOF take place in the building auditorium, which has no windows. The auditorium is protected from the building areas that have windows by internal walls. Therefore, the essential functions of the EOF are not expected to be adversely affected by a 100-year wind.

A description of the operation of the EOF and the EOF staff is contained in Chapter 2, Section 5.2 of the Trojan Radiological Emergency Plan. The alternate EOF is located at the PGE St. Helens District office in St. Helens, Oregon (13 miles from the Plant), and if activated will have the same staff and functions as the primary EOF. Emergency equipment and supplies are stored at the alternate EOF, and all EOF communications and dose assessment capabilities exist at the alternate EOF.

The alternate EOF will be prepared for activation whenever the EOF is activated by notifying the St. Helens District Manager. If evacuation of the EOF is required, the St. Helens office staff will be notified to begin activation of the alternate EOF. During the time required to move the EOF staff to the alternate EOF (approximately 30 minutes), the command, communications, and dose assessment functions of the EOF will be assumed by the TSC. The TSC staff is trained and qualified to carry on these tasks, and the required communications and dose assessment facilities exist in the TSC.

If the Emergency Response Manager (PGE Vice-President, Nuclear) has already reported to the EOF to assume command of the PGE emergency organization (Emergency Coordinator), he will turn his duties as Emergency Coordinator back to the Plant General Manager in the TSC until the alternate EOF is operational. State and county representatives at the EOF do not have decision-making roles, so relocation to the alternate EOF will not affect the emergency response capabilities of the State and county EOCs.

PGE will include anticipatory protective action criteria for relocation to the alternate EOF in the next amendment to the Trojan Radiological Emergency Plan which is expected to be issued in August 1981. These anticipatory protective action criteria for the EOF, which are shown below, should permit relocation prior to substantial releases of radioactivity from the Containment:

Fuel melting indicated by Containment area monitors or dose rate measurement outside Containment; and

1. Containment sprays and air coolers not functioning; and

2. Containment pressure greater than 70 psig for 2 min.; or

3. Other conditions exis which will lead to loss of Containment integrity.

The anticipatory protective action criteria are in addition to the following criteria which are based on radiation levels in the EOF:

1. One rem/hr whole body; or

2. 100 MPC I-131; or

3. Ten times these levels persisting for greater than 5 min.

PGE will test the relocation to and operation of the alternate EOF during the 1982 Trojan Emergency Plan Exercise.

HEPA filters are not required for the EOF ventilation system because of the protective action criteria established for the EOF. The anticipatory protective action criteria for the EOF (listed above) will require relocation to the alternate EOF before slow-developing accidents (i.e., greater than one hour) reach the point where significant particulate radioactivity will be released. For accidents involving a quick release of radioactivity (i.e., less than one hour) the releases will initially be dominated by volatile nuclides (noble gases and iodine). Particulate radioactivity will consist of iodine or short-lived noble gas daughters which do not contribute significantly to the dose rate inside the EOF. Furthermore, the presence of these short-lived noble gas daughters in the EOF would not be prevented by the HEPA filters because of their continuous generation by the decay of the noble gases inside the EOF. The dose rate and iodine air concentration protective action criteria will ensure relocation to the alternate EOF before significant long-lived particulate radioactivity such as Cesium-137 will be released.

IV. EOF DATA SYSTEM

The TSC data set discussed in Section II above will be transmitted to the primary EOF in the Visitors Information Center. As a minimum, both meteorological and radiological data will be transmitted to the alternate EOF in St. Helens, Oregon.

As shown in Table 1 the EOF data set contains the meteorological parameters required to calculate dispersion coefficients in the Evacuation Planning Zone (i.e., wind direction, wind speed, and 200-33 ft delta temperature) and Plant parameters required to calculate offsite doses (e.g., effluent radioactivity concentrations, effluent stream flow rates, area radiation monitor readings and Containment pressure). These data can be input along with radiation dose and airborne radioactivity measurements from field monitoring teams into an existing computer program which operates on a commercial time-share computer system located in Los Angeles, California. The computer can be accessed via commercial and PGE telephone circuits. Since PGE-owned microwave telephone circuits connect both the primary and alternate EOFs directly to major telephone exchanges in Portland, Oregon, access to the time-share computer will not be affected by the potential overload of telephone circuits in the vicinity of the Plant following an accident.

The computer program calculates atmospheric dispersion factors and offsite doses upon which offsite protective action recommendations can be made. The program incorporates terrain correction factors which compensate for the river valley terrain present in the Evacuation Planning Zone. By October 1, 1982 this program will be modified to calculate relative concentrations and offsite doses consistent with NUREG-0654, Appendix 2. PGE's course of action regarding the remaining items of NUREG-0654, Appendix 2 (i.e., a meteorological measurements system in accordance with proposed Revision 1 to Regulatory Guide 1.23 and remote interrogation of meteorological measurements) will be submitted to the NRC before January 1, 1982.

V. Safety Parameter Display System (SPDS)

The SPDS is an integral part of the TSC computer system and constitutes a series of graphic displays to provide the current status of the parameters important to safety in the main control room and the TSC command center. One of the graphic displays utilized for the SPDS is an iconic display of the critical parameters (Figures 4-3, -4, and -5 in WCAP-9725) in order to provide a quick recognition of accident conditions. Even with current technology, this display cannot be made available at the EOF without a great deal of difficulty and at a very high cost. The purpose of the iconic display is to provide an immediate identification and analysis of accidents in the control room and TSC. These functions are not performed by the EOF. Therefore, the iconic displays are not required in the EOF. Instead, the values of the critical parameters displayed in the iconic display along with their corresponding technical limits, will be transmitted to the EOF by the TSC Computer System. This will ensure that EOF personnel have the same data as that in the control room and TSC.

The TSC Computer System which develops the SPDS is not a safety system and thus, is not seismically qualified. In response to the NRC requirements in NUREG-0696 for a seismically qualified backup display system, the critical parameters listed in Table 2 will be displayed on a seismically qualified "post-accident" panel (panel C-09) which has been added to install the post-TMI instruments in the control room. Among the eight parameters listed in Table 2, the neutron flux indicators and steam generator water level recorders are displayed on panels C-02 and C-05, respectively. Although equipment qualifications for these parameters are still under evaluation for Regulatory Guide 1.97, selection of the parameters are based on ANS Standard 4.5 for accident recognition and mitigation.

The distances for detectability (i.e., trend reading of recorder and relative position of indicator) of recorder pointers and indicator pointers have been reviewed in a preliminary human engineering review. Detectability of meter pointers was subjectively evaluated and found to be adequate at 15 ft from the panel. However, detectability of recorder pointers was poor beyond 10 ft as was differentiation of recorder trace color. Since the distance between panels C-05 and C-09 is approximately 24 ft, the detectability distance of 10 ft for recorders from each panel leaves about 4 ft in which an operator needs to travel for adequate detectability. Obviously, detectability of the meters is not a problem. The distance of 4 ft (or 2 ft for each side) is insignificant compared to normal walking distance for the control room operators from panel to panel. The only two recorders for which an operator would have to move approximately 2 ft in either direction are steam generator water level on C-05 and Containment pressure on C-09. The three panels, C-02, -05 and -09, are thus considered to be adequately located for the purpose of a backup SPDS.

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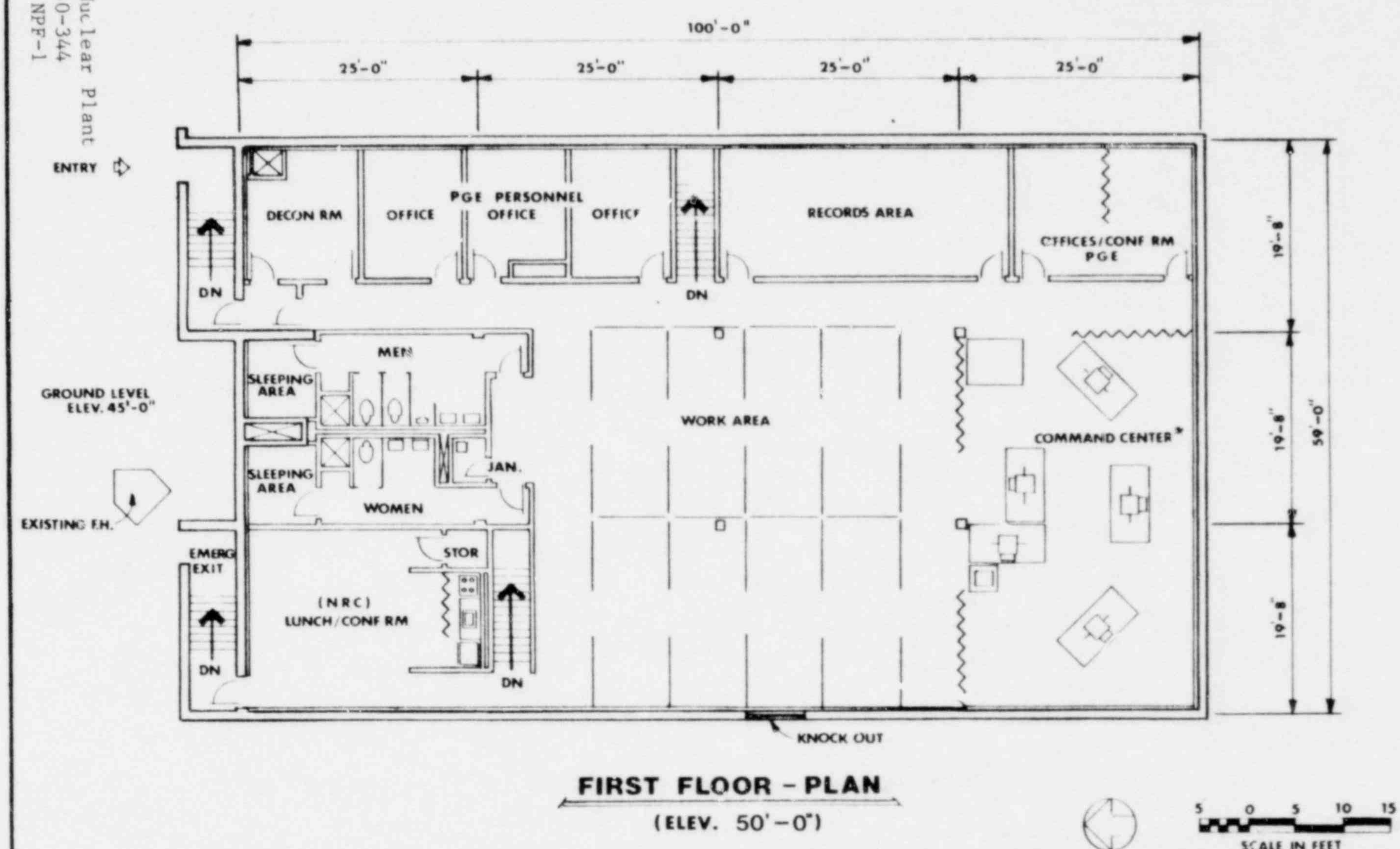

FIRST FLOOR - PLAN

(ELEV. 50'-0")

*REFER TO WCAP 9725 (JUNE 1980) FOR DETAILS

TROJAN NUCLEAR PLANT
PORTLAND GENERAL ELECTRIC

TECHNICAL SUPPORT CENTER

FIGURE 1

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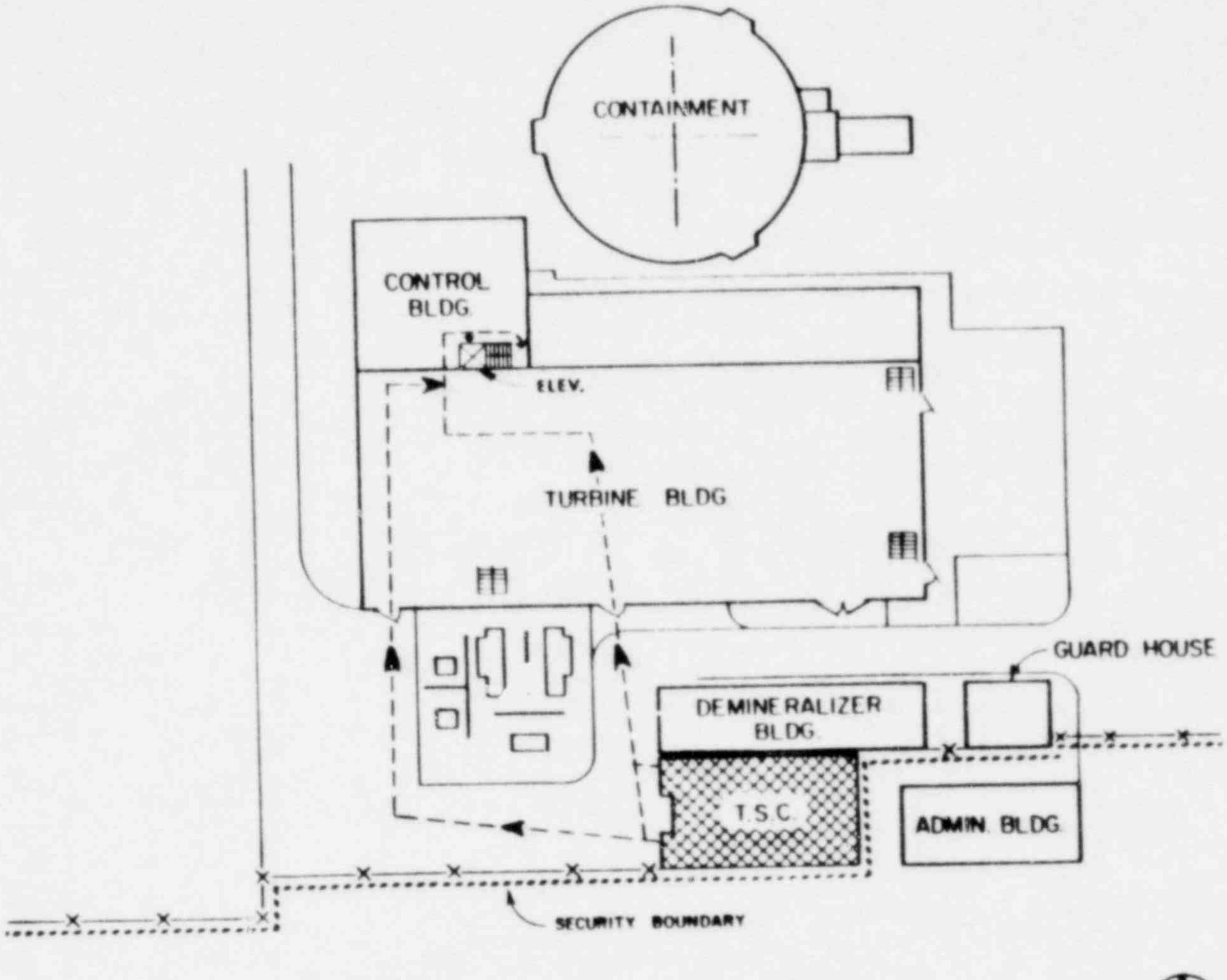

FIGURE - 2

EVACUATION ROUTE FROM TSC TO CONTROL ROOM

TECHNICAL SUPPORT CENTER
TROJAN NUCLEAR PLANT
PORTLAND GENERAL ELECTRIC

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Figure 3

Emergency Operations Facility
(Visitors Information Center)

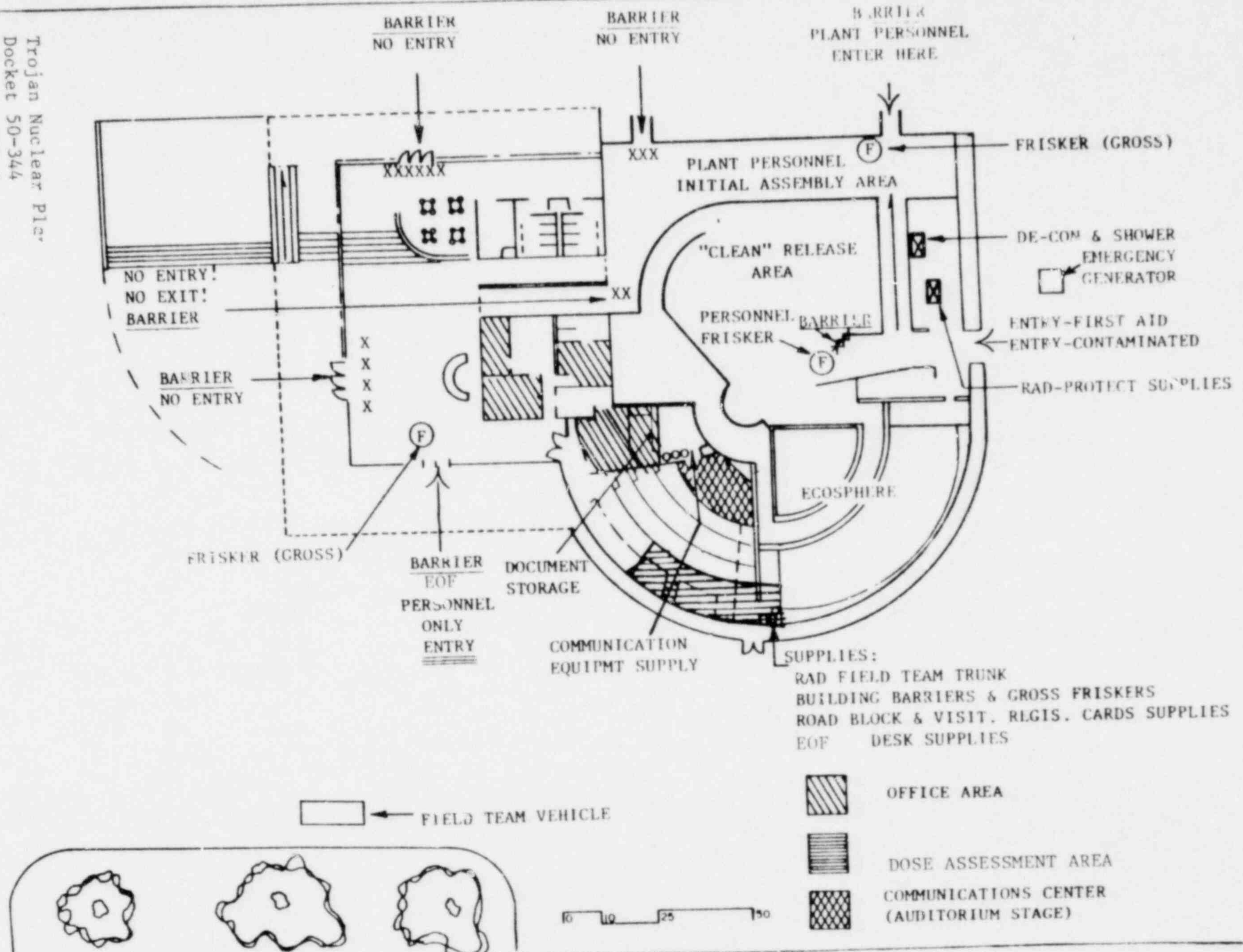

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TABLE 1

TROJAN NUCLEAR PLANT
TSC Data System P· ameters
(June 12, ·.)

1. SI Pump Discharge Flow
2. Charging Flow to BIT
3. Charging Flow
4. Letdown Flow
5. RCP Seal Injection Flow
6. Aux. Bldg. Vent Flow
7. Containment Purge Flow
8. Air Ejector Flow
9. Effluent Flow (Discharge and Dilution)
10. RCP Seal Leakage Flow
11. Emergency Boration Flow
12. RCS Flow
13. Steam Generator Feedwater Flow
14. Steam Generator Steam Flow
15. RHR Flow to Hotlegs
16. RHR Flow to Coldlegs
17. AFW Flow
18. Pressurizer Level
19. BAT Level
20. VCT Level
21. PRT Level
22. Steam Generator Level (narrow and wide ranges)
23. Accumulator Level
24. RWST Level

25. NAOH Tank Level

26. RCDT Level

27. Clean Waste Tank Level

28. Waste Monitor Tank Level

29. Dirty Waste Drain Tank Level

30. Dirty Waste Monitor Tank Level

31. CST Level

32. Containment Sump Level

33. Reactor Vessel Level

34. Emergency Diesel Tank Level

35. Diesel AFW Pump Tank Level

36. Wind Speed

37. Wind Direction

38. Dew Point Temperature

39. Delta Temperature

40. Containment Hydrogen

41. Nuclear Instrumentation System (Source, Intermediate and Power Ranges)

42. VCT Pressure

43. Charging Pump Discharge Pressure

44. Steam Generator Feedwater Flow Pressure

45. Nonregenerative HX Outlet Pressure

46. RCS Pressure

47. Pressurizer Pressure

48. PRT Pressure

49. SG Steam Pressure

TABLE 1

52. Accumulator Pressure
53. RHR Suction Pressure
54. Containment Spray Pump Discharge Pressure
55. Waste Gas Decay Tank Pressure
56. Turbine First Stage Pressure
57. Containment Pressure
58. Instrument Air Header Pressure
59. PRMs
60. ARMs
61. In-core Temperatures
62. Letdown HX Outlet Temperature
63. RCP Seal Water Temperature
64. RCP Lower Seal Water Bearing Temperature
65. RCS Hotleg Temperature
66. RCS Coldleg Temperature
67. Pressurizer Liquid Temperature
68. Pressurizer Steam Temperature
69. Pressurizer Relief Valve Discharge Temperature
70. PRT Temperature
71. RHR HX Inlet Temperature
72. Feed Water Temperature
73. RCS Delta Temperature
74. RCS T avg.
75. Tref
76. Containment Temperature

TABLE 1

77. CCW HX Inlet Temperature
78. CCW HX Outlet Temperature
79. RCP Motor Bearing Temperature
80. Containment Isolation Valve Positions (Open/Close)
81. Safety-Related System Valve Positions
82. Safety-Related Pump Status
83. ESF BUS/Breaker Status ($\geq$480 volt level)

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TABLE 2

TROJAN NUCLEAR PLANT
CRITICAL PARAMETERS FOR BACKUP SPDS ON POST-ACCIDENT PANEL (C-09)

Parameter	ANS Standard 4.5 Type B Function (Reg. Guide 1.97 Type B)	Panel Number	Installed?
1. Neutron Flux[a]	Reactivity Control	Operator Console C-02	Yes
2. Reactor Vessel Water Level	Reactor Core Cooling	C-09	Operational upon NRC Approval in 1982.
3. RCS Hotleg and Coldleg Temperatures[a]	Reactor Core Cooling	C-09[b]	Yes
4. RCS Pressure[a]	Reactor Core Cooling and RCS Integrity	C-09[b]	Yes
5. Containment Water Level	RCS and Containment Integrity	C-09	1/1/82
6. Containment Pressure (Recorder)[a]	RCS and Containment Integrity	C-09	1/1/82
7. Containment High-Range Area Radiation[a]	RCS Integrity	C-09	1/1/82
8. Steam Generator Water Level (Recorder)[a]	Not Included	C-05	Yes

(a) Seismic qualification is still under review and will be included in Regulatory Guide 1.97 evaluation.

(b) Information is available on demand through the subcooling margin monitor located on C-09.

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