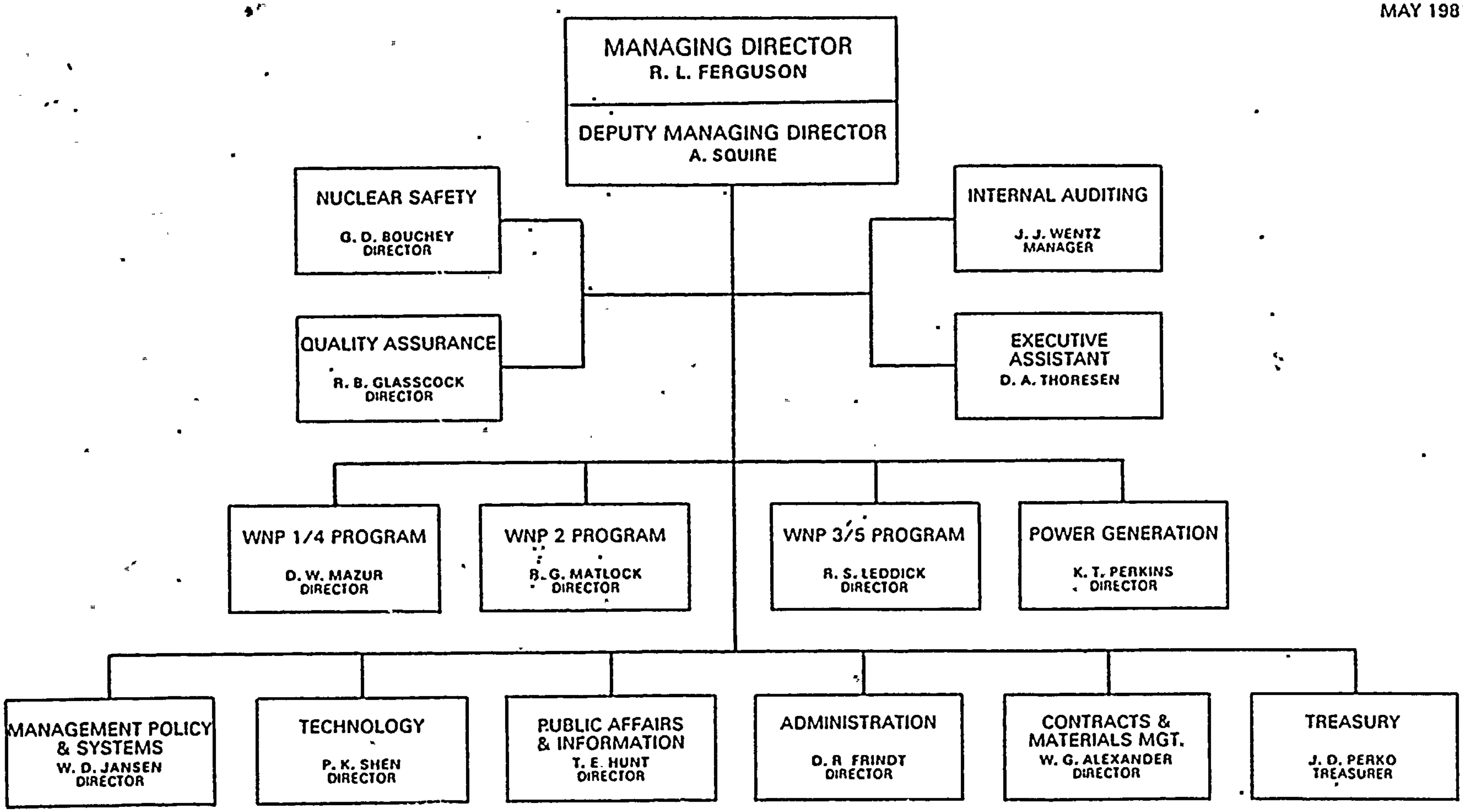
WASHINGTON PUBLIC POWER SUPPLY SYSTEM

MAY 1981



NOTE: THE CHIEF COUNSEL AND THE CONTROLLER, BOTH OF WHOM REPORT TO THE DIRECTOR, ADMINISTRATION, REPORT DIRECTLY TO THE MANAGING DIRECTOR IN AREAS INVOLVING ATTORNEY/CLIENT OR FISCAL/CLIENT RELATIONSHIP WITH THE MANAGING DIRECTOR AND/OR BOARD OF DIRECTORS.

8103114

ROBERT L. FERGUSON

POSITION: MANAGING DIRECTOR

EDUCATION: BS

EXPERIENCE:

1978 - 1980 DEPUTY ASSISTANT SECRETARY FOR U.S. DEPARTMENT OF ENERGY, NUCLEAR REACTORS PROGRAMS

- RESOURCE PLANNING
- MULTIPROGRAM MANAGEMENT
- REACTOR CONSTRUCTION PLANNING
- REACTOR OPERATIONS PLANNING

1973 - 1978 DIRECTOR, FAST FLUX TEST FACILITY, DOE

- NUCLEAR REACTOR CONSTRUCTION
- PROJECT MANAGEMENT
- OPERATIONS PLANNING

1971 - 1973 ASSISTANT MANAGER FOR PROGRAMS, RICHLAND AEC OFFICE

- FUEL DEVELOPMENT
- FUEL CYCLE
- NUCLEAR DEVELOPMENT

1961 - 1970 REACTOR SAFETY ENGINEER, CHICAGO OPERATIONS OFFICE AEC

- REACTOR SAFETY
- REACTOR OPERATIONS

ALEXANDER SQUIRE

POSITION: Deputy Managing Director
EDUCATION: BS
EXPERIENCE:

1971 - 1979 President, Westinghouse Hanford Corporation

- Reactor Design and Construction
- Reactor Development
- Fuel Development and Fabrication

1968 - 1971 Director of Purchasing and Traffic, Westinghouse Electric Corporation

- Procurement
- Logistics

1962 - 1967 General Manager, Plant Apparatus Division Westinghouse Electric Corporation

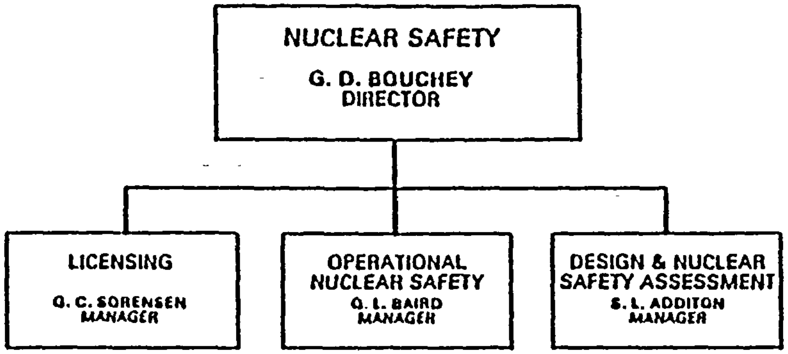
- Nuclear Equipment and Apparatus Design and Procurement
- Nuclear Systems Construction and Operation
- Control of Equipment and Manuals for Maintenance and Refueling of Naval Nuclear Reactors

1950 - 1961 Project Manager, Bettis Atomic Power Laboratory

- Development, Design, Management of Installation and Testing of Naval Nuclear Reactors
- Development of Original Materials, Welding, and Quality Control Specifications for Operating Reactors

WASHINGTON PUBLIC POWER SUPPLY SYSTEM

MAY 1981



G. DONALD BOUCHEY

POSITION: DIRECTOR, NUCLEAR SAFETY

EDUCATION: BSNE, MSNE, PHD

EXPERIENCE:

1975 - 1980 DIRECTOR, OPERATIONAL AND EXPERIMENTAL SAFETY DIVISION, FFTF PROJECT OFFICE

- o FFTF SAFETY AND LICENSING
- o FFTF CORE ENGINEERING

1972 - 1975 REACTOR ENGINEER/SYSTEMS ENGINEER, U. S. ENERGY RESEARCH & DEVELOPMENT AGENCY

- o FFTF DESIGN

1967 - 1972 SUPERVISOR, NUCLEAR REACTOR LABORATORY, UNIVERSITY OF TEXAS, AUSTIN, MECHANICAL ENGINEERING DEPARTMENT

- o RESEARCH
- o LICENSED SRO
- o RADIATION DETECTION AND HEALTH PHYSICS
- o TEACHING

INFORMATION ON NUCLEAR SAFETY MANAGEMENT AND STAFF

DEPARTMENT: NUCLEAR SAFETY

AS OF: 8/31/81

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
G. D. Bouchey	BS Nuclear Eng. MS Nuclear Eng. PhD	Director, Nuclear Safety	10 mo./17 yr./18 yr.	Safety & Licensing Core Engineering Design	Research Radiation Detection Health Physics Research/Teaching	U. S. Dept. of Energy (FFTF)	Director, Operational & Experimental Safety Div.
S. E. Bussman	None	Administrative Assistant	1 yr./2 yr./3 yr.	Administration	Secretarial	Peppermill, Inc.	Office Manager

+Registered P.E., Washington.
✓Registered P.E., other state(s).

INFORMATION ON NUCLEAR SAFETY MANAGEMENT AND STAFF

DEPARTMENT: LICENSING

AS OF: 8/31/81

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
G. C. Sorensen	BS Mechanical Engineering	Manager, Licensing	8 yr./7 yr./15 yr.	Licensing	Mechanical Engineering	Amphenol Sams Division	Project Engineer
R. M. Nelson	BS Mechanical Engineering Nuclear Option	Manager, WNP-2 Project Licensing	2 mo./21 yr./21 yr.	Nuclear Eng. Mech. Eng. Design	NSSS Design NSSS Licensing Chemical Eng.	General Electric Co.	Senior Licensing Engineer
A. G. Hosler	BS Electrical Eng. MS Nucl. Eng.	Manager, WNP-1/4 Priject Licensing	5 mo./18 yr./18 yr.	Licensing Nucl. Eng.	Analysis	Westinghouse	Senior Development Engineer
✓ K. W. Cook	BS Mechanical Eng. Nuclear Option MS Mechanical Eng. ABC Courses (GE)	Manager, WNP-3/5 Project Licensing	16 mo./20 yr./21 yr.	Nucl. Eng. Licensing	Mech. Eng. Safety Eval.	General Electric Co.	Senior Licensing Engineer
K. A. Hadley	BS Mechanical Eng. BS Industrial Eng. Naval Nuclear Power School	Engineer I	1 mo./15 yr./15 yr.	Mechanical Eng. Industrial Eng.	Operations/Maint./ Systems Generic Issues	Westinghouse	Cognizant Design Engr.
✓ R. G. Joshi	BS Electrical Eng. MS Nuclear Eng.	Senior Engineer	2 mo./14 yr./14 yr.	Licensing Eng. Nuclear Eng.	I&C Eng. Electrical Eng.	Stone & Webster Energy Corp.	Licensing Engineer
P. L. Powell	BS Electrical Eng. Naval Nuclear Power School	Engineer I	4 yr./7 yr./11 yr.	Nuclear Eng. Electrical Eng.	Safety Analysis Administrative Eng.	Westinghouse	Admin. Engineer
A. J. Moore	BS Physics Grad. Studies in Meterology	Project Engineer I	2 yr./12 yr./14 yr.	Operations at FFTF (Operator Training) NDE/NDT Training Lead Auditor	Quality Assurance	Westinghouse	Reactor Operator

+ Registered P.E., Washington.
✓ Registered P.E., other state(s).

INFORMATION ON NUCLEAR SAFETY MANAGEMENT AND STAFF

DEPARTMENT: LICENSING

AS OF: 8/31/81

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
C. D. Taylor	AA	Licensing Assistant, WNP-2	8 yr./ /8 yr.	FSAR Amendments	NRC Questions	------	------

+Registered P. E., Washington.
√Registered P. E., other state(s).

INFORMATION ON NUCLEAR SAFETY MANAGEMENT AND STAFF

DEPARTMENT: OPERATIONAL NUCLEAR SAFETY

AS OF: 8/31/81

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
Q. L. Baird	AB Mathematics MS Physics PhD Physics	Manager, Operational Nuclear Safety	6 mo./23 yr./24 yr.	Nuclear Eng. Reactor Physics	Nucl. Safety Tech. Reactor Operations	Westinghouse	Manager, Operational Nuclear Sfty
C. H. McGilton	MS Metallurgy ORNL School of Reactor Tech. Licensed SRO on Cooper	Manager, SEG WNP-2	8 yr./10 yr./18 yr.	Reactor Operations Operations Supv. at WNP-2 (7 yr.)	Nuclear Eng.	Northern States Power	Operations Engineer
S. C. Denison	BS Mechanical Eng.	Manager, SEG WNP-1/4	8 yr./9 yr./17 yr.	Nuclear Eng. Mechanical Eng.	Reactor Operations	Westinghouse	Senior Proj. Engineer
D. W. Coleman	BS Mechanical Eng. MS Mechanical Eng. Grad. work in Nuclear Eng.	Manager, SEG WNP-3/5	6 mo./12 yr./13 yr.	Mechanical Eng.	Nuclear Eng.	U. S. Dept. of Energy	Nuclear Eng.
F. D. Frisch	BS Mechanical Eng. Licensed SRO at Millstone 1 & Pilgrim	Principal Engineer	6 yr./15 yr./21 yr.	Mechanical Eng. Reactor Operations	Start-Up Testing	General Electric Co.	Start-Up Operations Super-intendent
R. G. DaValle	BS Engineering	Senior Engineer	8 yr./7 yr./15 yr.	Reactor Operations Control Room Human Factors Review	General Eng.	Bechtel Power	Sr. Field Scheduling Engineer
A. K. Yee	BS Physics MS Physics PhD Physics Grad. work in Nuclear Eng.	Principal Engineer	4 mo./31 yr./31 yr.	Nuclear Eng. Reactor Analysis Safety Analysis	Nuclear Safety Tech.	Westinghouse	Principal Eng

+Registered P. E., Washington.
✓Registered P. E., other state(s).

INFORMATION ON NUCLEAR SAFETY MANAGEMENT AND STAFF

DEPARTMENT: OPERATIONAL NUCLEAR SAFETY

AS OF: 8/31/81

	NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
✓	M. D. Zentner	BS Physics MS Nuclear Eng.	Senior Engineer	1 mo./8 yr./8 yr.	Nuclear Eng. Physics	Reactor Analysis Reactor Operations	Brookhaven National Lab.	Research Engineer
+	R. M. Norris	BS Mechanical Eng. MS Mechanical Eng.	Principal Engineer	5 yr./10 yr./15 yr.	Mechanical Eng.	Nuclear Eng. NSSS Design	General Electric Co.	Program Manager

+ Registered P. E., Washington.
✓ Registered P. E., other state(s).

INFORMATION ON NUCLEAR SAFETY MANAGEMENT AND STAFF

DEPARTMENT: DESIGN AND NUCLEAR SAFETY ASSESSMENT

AS OF: 8/31/81

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
S. L. Additon	BS Mathematics MS Nuclear Eng. BA	Manager, Design & Nuclear Safety Assessment	7 mo./13 yr./14 yr.	Thermal Hydraulic & Plant Dynamic Analysis	Nuclear Safety	Westinghouse	Manager, Systems Design
✓ J. W. Sale	BS Mechanical Eng MS Mechanical Eng Reactor Operation	Manager, Desing Assessment	3 yr./18 yr./21 yr.	Mechanical Eng.	BOP System Design Fire Protection	Pacific Gas & Electric Co	Mechanical Engineer
R. O. Vosburgh	BA Physics/Math MS Physics	Manager, Nuclear Safety Analysis	1 yr./16 yr./17 yr.	Nuclear Eng. Transient Anal. (T-H Neutronic)	Risk Assessment	Babcock & Wilcox	Senior Engineer
S. M. Baker	BS Mathematics MS Nuclear Eng. PhD Mathematics	Manager, Nuclear Safety Standards & Technology	7 mo./16 yr./17 yr.	Mathematics Nuclear Eng.	Reactor Operations	U. S. Dept of Energy-FFTF	Safety & Operations Staff
C. J. Foley	BS General Eng. MS Nuclear Eng.	Principal Engineer	7 mo./15 yr./16 yr.	Mechanical Eng.	Nuclear Plant Start-Up Supv. (1975-1980)	Westinghouse	Principal Engineer
✓ R. A. Hazen	None	Senior Engineer	6 mo./19 yr./20 yr.	BWR Systems	BOP	General Electric	Senior Project Eng.
E. D. Shoua	BS Electrical Eng. MS Structural Eng. PhD Material & Applied Mech.	Senior Engineer	4 yr./11 yr./15 yr.	Nuclear Fuel Safety Analysis	Material & Stress Analysis Structural Design	Nuclear Services Corp.	Nuclear Fuel Consultant
+ J. M. Henderson	BS Nuclear Eng.	Engineer I	1 mo./9 yr./9 yr.	Nuclear Eng. Instrumentation	Computer Systems Nuclear Fuel	Westinghouse	Engineer I

+Registered P.E., Washington.
✓Registered P.E., other state(s).

INFORMATION ON NUCLEAR SAFETY MANAGEMENT AND STAFF

DEPARTMENT: DESIGN AND NUCLEAR SAFETY ASSESSMENT AS OF: 8/31/81

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
P. R. Shire	BS Math/Chem/ Physics MS Nuclear Eng.	Principal Engineer	5 mo./27 yr./27 yr.	Nuclear Eng. LWR/LMFRB Safety Analysis Neutronics/Thermal Hydraulics Ana	Plant Heat Balance Generation Planning Computer Code Spec.	Westinghouse	Reactor Safety Eng.
J. S. Dukelow	AB Mathematics MA Mathematics MS Nuclear Eng.	Senior Engineer	4 mo./11 yr./11 yr.	Nuclear Eng. Thermal Hydraulics	Reliability Mathematics	HEDL	Reliability Engineer
D. L. Prezbindowski	BS Engr. Science MS Nuclear Eng. PhD Nuclear Eng.	Senior Engineer	5 mo./13 yr./13 yr.	Mechanical Eng. Nuclear Eng.	Core Physics & Shielding Calculations	Battelle	Sr. Patent Engineer
✓ B. L. Twitty	BS MS Radiochemistry MS Nuclear Eng. Certified in Health Physics	Principal Engineer	3 mo./25 yr./25 yr.	Reactor Control Inst. Startup - Accept.	Safety Anal., Health Physics, Plant Design, & Radiochemistry	Westinghouse	Senior Engineer

+ Registered P. E., Washington.
✓ Registered P. E., other state(s).

WASHINGTON PUBLIC POWER SUPPLY SYSTEM

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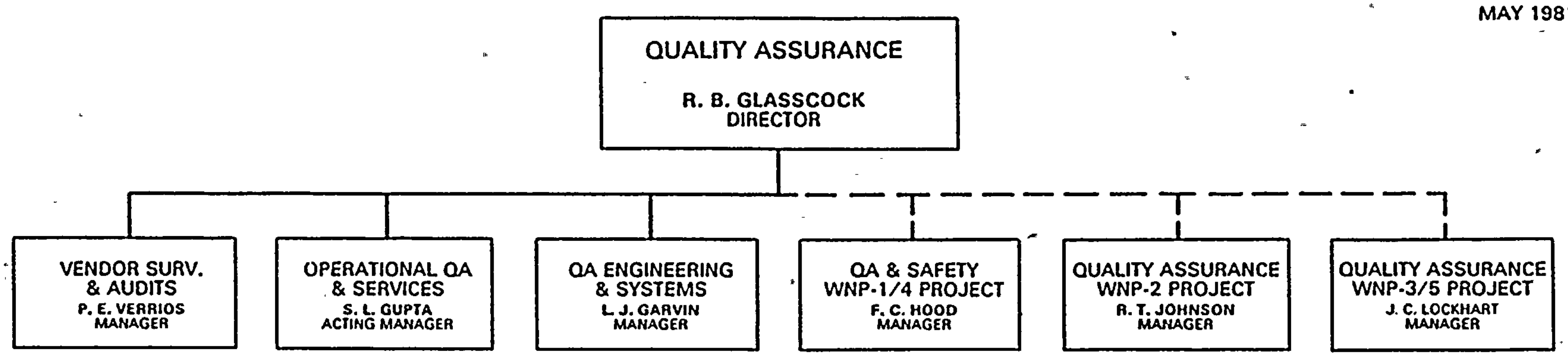

8105041

ROBERT B. GLASSCOCK

POSITION: DIRECTOR, QUALITY ASSURANCE

EDUCATION: BSEE, PE

EXPERIENCE:

1976 - 1981 DIRECTOR, QUALITY ASSURANCE, D.O.E. FFTF PROJECT OFFICE

- CONSTRUCTION
- ACCEPTANCE TESTING
- OPERATION
- RAND D PROGRAMS

1972 - 1976 QUALITY ASSURANCE SPECIALIST, ENFORCEMENT DIVISION, NUCLEAR REGULATORY COMMISSION

- CONSTRUCTION
- OPERATION

1958 - 1972 MANAGER, PROJECT QUALITY ASSURANCE, AEROJET GENERAL CORPORATION

- CONSTRUCTION
- OPERATION

1955 - 1958 MANAGER CONTROL, ACF INDUSTRIES, BUFFALO PLANT

- QUALITY CONTROL DIRECTION

1950 - 1955 CHIEF, QUALITY CONTROL, AEC KANSAS CITY FIELD OFFICES

- QUALITY CONTROL PLANNING AND DIRECTION

INFORMATION ON QUALITY ASSURANCE MANAGEMENT AND STAFF

DEPARTMENT: Quality Assurance

AS OF: September 8, 1981

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
Casavant, R. D.	(A.S.) Business Administration	QA Engineer I	3 mos/2.5 yrs/2.75 yrs.	Mechanical Systems	Business Admin.	Crosby Valve & Guage Co.	QA Supvr.
✓Chandon, H. C.	(B.S.) Electrical Engineering	Senior QA Engineer	3.5 yrs/19.5 yrs/ 23 yrs	QA Engineer	Electrical Engineer	Bechtel Power Corp.	Sr. QA Engr.
✓Dunn, T. E.	(B.S.) Mechanical Engineering	Principal QA Engineer	1 mo/22 yrs/22 yrs	QA Program Development & Maintenance	Fuel Fabrication QA Programs	Battelle	Sr. Scientist
Feldman, D. S.	(B.S.) Industrial Engineering	QA Engineer I	1 yr/5 yrs/6 yrs	QA Operations	Power Plant Operations	U. S. Navy	QA Engineer
✓Garvin, L. J.	(B.S.) Chemical Engineering (B.S.) Nuclear Engineering) (B.S.) Metal. Engineering	Manager, Quality Performance & Measurements	2 yrs/17 yrs/19yrs	QA & Metallurgical Engineering	Nuclear Safety	USNRC	Reactor Inspector
Gross, V. R.		QA Engineer I	4 yrs/14 yrs/18 yrs	Mechanical/Elect.	I/C, Civil	UE&C	QA Supervisor
Gupta, S. L.	(B.S.) Physics	Principal QA Engineer	2.5 yrs/17.5 yrs/ 20 yrs	Quality Assurance	a. Fueling & Refueling Operations b. Nuclear Plant Operations	Carolina Power & Light	Senior QA Engineer
Henthorn, C. O.		Senior QA Engineer	1.5 yrs/31.5 yrs/ 33 yrs	Quality Assurance	Business Management	Self	QA Consultant
Houchins, T. J.		Manager, Audits and Surveillance	7 yrs/19 yrs/26 yrs	Quality Assurance	Auditing	UE&C	Site Surveillance Supervisor

† Registered P. E., Washington.
✓ Registered P. E., other state(s).

INFORMATION ON QUALITY ASSURANCE MANAGEMENT AND STAFF

DEPARTMENT: Quality Assurance

AS OF: September 8, 1981

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
Kooy, W.	(B.S.) Electrical Engineering	Principal QA Engineer	4.5 yrs/16.5 yrs/ 21 yrs	Test & Startup Electrical, Instrument, & Control	Chemical Analysis	MATSCO	Director of Containment Testing
✓Krolicki, R. P.	AA-Radiogr. BA-Commerce MA-Economics	Principal QA Engineer	4 yrs/23 yrs/27 yrs	Quality Assurance, NDE	Welding	Florida Power & Light Co.	Senior QA Engineer
Lauck, J. A.	(B.S.) Accounting	QA Engineer I	1 mo/12 yrs/12.1 yrs	Mechanical-equip./ piping erection & testing	Welding/NDE	WBG #2	Procurement Engineer
Simons, R. M.		Manager, Quality Systems	1.5 yrs/22 yrs/ 23.5 yrs	QA Systems	NDE	Richland Engineering	QA Manager
Walker, J. M.		Senior QA Engineer	6 yrs/5.5 yrs/11.5 yrs	QA Surveillance/ Audits	QA Systems/Programs	Northeast Utilities Service Co.	Engineering Technician
Webster, R. D.	(BA) Education	QA Engineer I	6 mos/39.5 yrs/ 40 yrs	Systems Mechanical	Electrical	Westinghouse	QA Engineer
Winters, H. L.		QA Receiving Inspector	6 yrs/21 yrs/27 yrs	Receiving Inspection	Mechanical, Dimensional Measurements	HUICO	QA Inspector
✓Wooley, G. H.		Manager, Vendor QA	4.5 yrs/5.5 yrs/ 10 yrs	QA General Administration, Supervisory	Mechanical, Welding, NDE	Bechtel	Vendor Surveillance Engineer

+Registered P. E., Washington.
✓Registered P. E., other state(s).

INFORMATION ON QUALITY ASSURANCE MANAGEMENT AND STAFF

DEPARTMENT: Quality Assurance

AS OF: September 8, 1981

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
✓Zimmerschied, J. R.	(B.S.) Physics Engineering	Senior QA Engineer	3 yrs/9 yrs/12 yrs	QA Surveillance & Audits in piping & mechanical area	Programming and use of micro-processors	Bechtel Power Corp.	Project QA Engineer

+Registered P. E., Washington.
✓Registered P. E., other state(s).

WASHINGTON PUBLIC POWER SUPPLY SYSTEM

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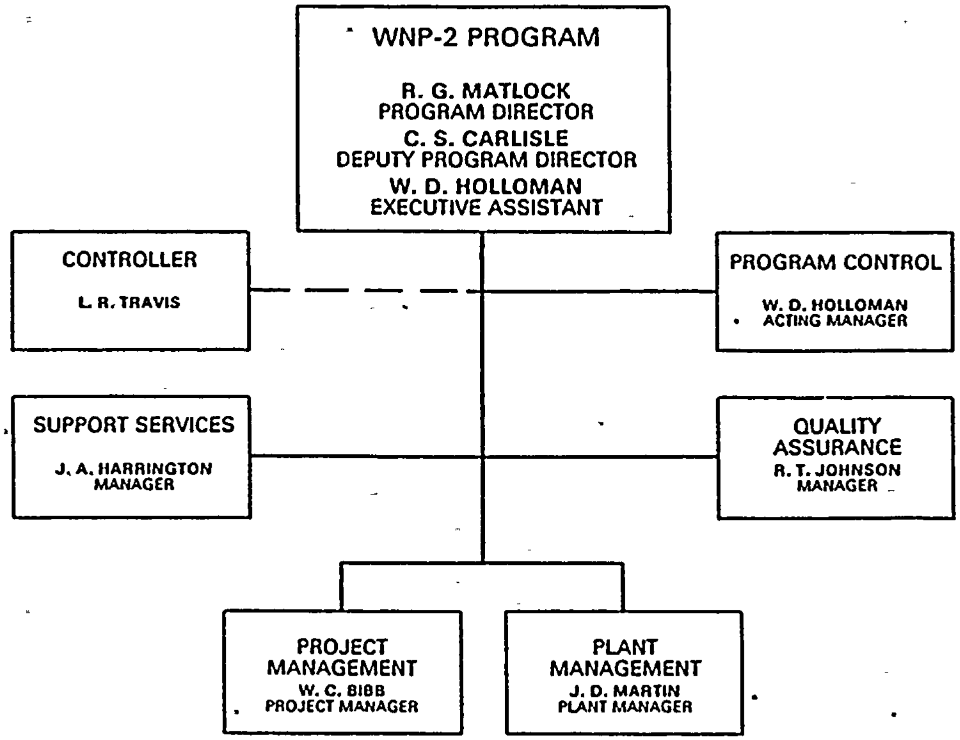


8105041

ROBERT G. MATLOCK

POSITION: PROGRAM DIRECTOR, WNP-2

EDUCATION: BSME, PHD PHYSICS

EXPERIENCE:

1979 - 1980 DEPUTY DIRECTOR OF NUCLEAR REACTOR PROGRAMS FOR THE U.S. DEPARTMENT OF ENERGY

- RESOURCE PLANNING
- MULTIPROGRAM MANAGEMENT
- OPERATIONS PLANNING

1966 - 1979 ARGONNE NATIONAL LABORATORY

77 - 79 DIRECTOR OF FOSSIL ENERGY AND SOLAR ENERGY PROGRAMS

- PROGRAM DEVELOPMENTS
- ENERGY RESOURCE ANALYSIS
- LABORATORY WIDE TECHNICAL PROGRAM MANAGEMENT

74 - 79 PROJECT MANAGER FOR THE FAST REACTOR SAFETY TEST FACILITY PROJECT

- SPECIFICATION AND REQUIREMENTS DEVELOPMENT
- DESIGN DEVELOPMENT
- CONTRACT MANAGEMENT
- SAFETY EVALUATION AND ENVIRONMENTAL REPORT PREPARATION

73 - 74 EXECUTIVE ASSISTANT TO THE DIRECTOR OF REACTOR PROGRAMS

- LABORATORY WIDE PROGRAM PLANNING AND MANAGEMENT
- FACILITIES SAFETY REVIEW AND ASSESSMENT

71 - 73 MANAGER ANALYSIS AND TEST PROGRAM ON EBR-II

- ONGOING OPERATIONAL SAFETY ANALYSIS
- TECHNICAL SPECIFICATION PREPARATION
- SAFETY ANALYSIS REPORT REVISIONS
- EXPERIMENT REVIEW AND APPROVAL

66 - 71 FACILITY MANAGER/OPERATIONS MANAGER ON THE ZERO POWER PLUTONIUM REACTOR (ZPPR)

- FACILITY CONSTRUCTION COMPLETION
- SAFETY ANALYSIS REPORT PREPARATION
- OPERATING MANUALS AND TECHNICAL SPECIFICATION PREPARATION
- STAFFING AND TRAINING
- FACILITY STARTUP AND OPERATION

INFORMATION ON WNP-2 MANAGEMENT AND STAFF

DEPARTMENT: WNP-2 PROGRAM DIRECTOR

AS OF: 9-1-81

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
R.G. Matlock	BSME PH.D. Physics	Program Director, WNP-2	1 yr/20 yr/21 yr	Nuclear Reactor Design, Operations, Project Management	Nuclear Physics, Research & Development.	U.S. DOE, ANL	Deputy Director, for Nuclear Programs, Physicist, Program Director, Project Mgr

+ Registered P. E., Washington.
✓ Registered P. E., other state(s).

INFORMATION ON WNP-2 MANAGEMENT AND STAFF

DEPARTMENT: WNP-2 PROGRAM DIRECTOR

AS OF: 9/1/81

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
C.S. Carlisle	BS Naval Science MS Business Admin	Deputy Program Director, WNP-2	4 mos/35 yrs/35 yrs	Nuclear plant operations and maintenance	Construction Project Management	U.S. Navy	25 yrs experience in nuclear submarine construction, operations, & maintenance.

+Registered P. E., Washington.
✓Registered P. E., other state(s).

DEPARTMENT: PROGRAM CONTROL

AS OF: September 4, 1981

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
W.D. Holloman	B.S. Eng. (Gen) (U.S. Naval Academy) B.S. Eng. (M.E.) Nuclear Power	Executive Assistant Mgr. Program Control (Acting)	6 mos/26yrs/26yrs	Resource Mgmt.	Tech Management (Nuclear Pwr R&D)	AEC/DOE 6 yrs U.S. Navy 20 years	(Budget Officer) (Line Officer)

+Registered P. E., Washington.
√Registered P. E., other state(s).

INFORMATION ON WNP-2 MANAGEMENT AND STAFF

DEPARTMENT: Programs's Directors

AS OF: September 4, 1981

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
J.R. Honekamp	B.S. & M.S. Chem Eng. PH.D Chem Eng. with minor in Nuclear Eng.	Technical Specialist	10 mo./20/21	Design and Testing of Nuclear Power Plants	Fuels and Materials Performance, and Operations	Argonne National Lab and Knolls Atomic Power Lab.	11 years experience in design and testing associated with Breeder Reactor Program 9 years experience in design, testing and operation of Naval Power Plants

+Registered P. E., Washington.
✓Registered P. E., other state(s).

INFORMATION ON WNP-2 MANAGEMENT AND STAFF

DEPARTMENT: Project Quality Assurance

AS OF: 9-4-81

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
Roger T. Johnson	None	WNP-2 Project QA Manager	WPPSS 8 years QA Mngmt. functions. 9 years experience in Nuclear Const. and Project Mngmt. 17 years total experience.	Nuclear Const. and Project Mgmt.	Mechanical systems design, installation and inspection.	Westinghouse Hanford	Project Engineer
Richard E. Spence	B.S. Nuclear Engineering	QA Compliance Supervisor	1½ yrs/14yrs/15yrs	Nuclear power plant const. and Quality Assurance.	Nuclear power plant test, startup and operations. Also, radiation protection	Mid-Columbia Engineering Co.	Project QA Engineer
Carl O. Wright	BSME	QE Supervisor	3yrs/7½yrs/10½yrs	Nuclear power plant const. Quality Assurance Audits/Surveillances	Mechanical Engineering with power utilities.	Bechtel 4½yrs.	SR QAE

+Registered P.E., Washington.
✓Registered P.E., other state(s).

INFORMATION ON WNP-2 MANAGEMENT AND STAFF

DEPARTMENT: OPERATIONAL QUALITY ASSURANCE

AS OF: 9/8/81

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
Joseph M. Graziani	B.S. Materials Engineering, 6/71	Operational QA Supervisor WNP-2	2 yrs 10 mo. w/Supply System 6 yrs 4 mo w/Bechtel 9 yrs. 2 mo - Total	Nuclear Construction Field Engineer, (startup System completion & Electrical check-out)	Operational Quality Assurance Engineer -Nuclear Management Plan for PRE-OP and operation phase	Bechtel Power Corp.	Electrical Field Engineer Startup System Turnover Administrator Construction Completion Coordinator Lead Field Cost Engineer Field Cost Engineer

Registered P. E., Washington.
Registered P. E., other state(s).

INFORMATION ON WNP-2 MANAGEMENT AND STAFF

DEPARTMENT: SUPPORT SERVICES

AS OF: September 1, 1981

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
J. A. Harrington	BA Business Administration	Manager, Support Services Division	7 yrs/18 yrs/25 yrs	Business Management	Quality Assurance Management	Bunker-RAMO Corporation	Reliability & Quality Control Manager

+ Registered P. E., Washington.
✓ Registered P. E., other state(s).

INFORMATION ON WNP-2 MANAGEMENT AND STAFF

DEPARTMENT: Administration

AS OF: September 3, 1981

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
J. F. Peters	BA Psychology	Administration Manager WNP-2	5 yrs./6 yrs./11 yrs.	Commercial Nuclear Plant Administration & Support Services	Operations and Startup Management	Metropolitan Edison Co.	Station Administrator

+Registered P. E., Washington.
✓Registered P. E., other state(s).

INFORMATION ON WNP-2 MANAGEMENT AND STAFF

DEPARTMENT: WNP-2 RECORDS MANAGEMENT

AS OF: September 1, 1981

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
T.V. Anderson	BA History	Manager, Records Management	4yrs/3 yrs/7 yrs	Records Management	Archives Administration	National Archives & Records	Records Management Intern

+Registered P. E., Washington.
√Registered P. E., other state(s).

INFORMATION ON WNP-2 MANAGEMENT AND STAFF

DEPARTMENT: Project Materials Management

AS OF: September 1, 1981

NAME	DEGREE	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPLE AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
D. R. Herling	AA Business Admin.	Manager, Project Materials Management, WNP-2	4 yrs/15 yrs/19 yrs	Procurement & Materials Mgmt. Process Plants & Nuclear Construction	Manufacturing	Davy McKee	Project Material Coordinator

INFORMATION ON WNP-2 MANAGEMENT AND STAFF

DEPARTMENT: Business

AS OF: September 1, 1981

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
R. E. Baker	BS Management & Military Science MS Mech. Engineering	Business Manager	4½ yrs/23 yrs/23 yrs	Business	Construction Mgmt.	CSTDC/Great American Ins. Company	Consultant

+ Registered P. E., Washington.
✓ Registered P. E., other state(s).

INFORMATION ON WNP-2 MANAGEMENT AND STAFF

DEPARTMENT: OPERATIONS

AS OF: 9/4/81

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
J. D. MARTIN	BS Industrial Management Brigham Young 1960	WNP-2 Plant Manager	3 / 18 / 21	Reactor Operations	BWR Startup	US NRC	Reactor Inspector
						General Elec. Browns Ferry	Ops. Mgr. Ops. Supt. Training Inst. Startup Test. Engr.
						GE-Hanford	Reactor Ops Supervisor
						DUN	Reactor Oper Supervisor

+Registered P. E., Washington.
√Registered P. E., other state(s).

INFORMATION ON WNP-2 MANAGEMENT AND STAFF

DEPARTMENT: TRAINING

AS OF: 9/4/81

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
R. D. DAVIDSON	BS in Science Education	WNP-2 Training Supervisor	6 / 9 / 15	Development and Managing Training Programs	Reactor Operation	General Elect.	Simulator Training Engineer
						US Navy	Reactor Oper/ Training

+Registered P. E., Washington.
√Registered P. E., other state(s).

INFORMATION ON WNP-2 MANAGEMENT AND STAFF

Page _____ of _____

DEPARTMENT: Plant Technical

AS OF: September 4, 1981

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
K. D. Cowan Registered Prof. Engineer, Mech.-California	1) BS-Mech. Eng. 2) Masters-Bus. Administration	WNP-2 Plant Technical Superintendent	WPPSS-WNP-2 Project Engineering Manager 1½ years WPPSS-WNP-2 Plant Technical Supt. 1 year GE-Nuclear Energy Division 1) Project Eng. 2) Systems Eng. 3) Design Eng. 17 years total	Technical or Project Engineering Management	Mechanical Systems Engineering	General Electric - Nuclear Energy Division	Various-from Design Engineer to Project Engineer to Systems Engineer to Unit, Subsection and Section Manager

+ Registered P. E., Washington.
√ Registered P. E., other state(s).

INFORMATION ON WNP-2 MANAGEMENT AND STAFF

DEPARTMENT: Operations/Maintenance

AS OF: September 3, 1981

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
R. L. Corcoran	B.S.E.E.	Plant Operations Superintendent	8/7/15	Nuclear Plant Operations	Plant Technical/ Engineering	Dairyland Power Coop. LaCrosse BWR Gen. Station	Plant Tech. Supervisor/ Project Engr
J. W. Hedges	B.S.M.E. M.S. Geology	Plant Maintenance Supervisor	8/23/31	Nuclear Plant Maintenance	Plant Technical/ Engineering	UNI, N-Reactor	Sr. Engr. N-Reactor Maintenance

+Registered P. E., Washington.
✓Registered P. E., other state(s).

INFORMATION ON WNP-2 MANAGEMENT AND STAFF

DEPARTMENT: Project Management

AS OF: 9/4/81

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
W. C. Bibb	3 yrs towards EE degree Registered P.E., California	Project Manager, WNP-2	5 yrs/22 yrs/ 27 yrs	Startup/Ops/Power Production Mgr	Project Management	General Elect.	Area Mgr Nuclear Services Midwest Reg. Site Mgr Cooper Nuc. Station

+Registered P. E., Washington.
✓Registered P. E., other state(s).

INFORMATION ON WNP-2 MANAGEMENT AND STAFF

Page _____ of _____

DEPARTMENT: Test & Startup

AS OF: ______________

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
G. K. Afflerbach	BS Electrical Engineering	Deputy Project Manager, Startup, WNP-2	3 yrs/14.5 yrs/17.5 yrs	Nuclear plant startup,operation and maintenance	Nuclear plant retrofit & outage management	General Electric	Operations Manager

+Registered P. E., Washington.
√Registered P. E., other state(s).

INFORMATION ON WNP-2 MANAGEMENT AND STAFF

DEPARTMENT: Construction Management

AS OF: September 4, 1981

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
H.A. Crisp	B.S. Mining Eng. MS Petroleum Eng. MS Civil Eng.	Deputy Project Manager Construction	1 yr/25yrs/26yrs	Management of Plant Construction and Maintenance	Engineering	U.S. Navy	25 years experience in facilities maintenance and const. last 7 yrs. associated with Nuclear or Nuclear Support Facilities

+Registered P. E., Washington.
✓Registered P. E., other state(s).

INFORMATION ON WNP-2 MANAGEMENT AND STAFF

DEPARTMENT: WNP-2 Project Engineering 62900

AS OF: 9/4/81

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
BA Holmberg	BS Marine Engrg.	Deputy Project Manager-Engineering	2.75/20/22.75	Management/Nuclear/Operations	Mechanical	U.S. Navy	Commander/Navy

+Registered P. E., Washington.
√Registered P. E., other state(s).

INFORMATION ON WNP-2 MANAGEMENT AND STAFF

DEPARTMENT: Project Finance

AS OF: September 8, 1981

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
L.R. Travis	B.S. Business Administration	Project Controller, WNP-2	3 yrs./31 yrs./34 yrs.	Construction Financial Management	Manufacturing Financial Management	Kaiser Engineers	Administrative Manager
C.W. Robinson	B.S. Commerce & Finance	Project Accounting Manager, WNP-2	BRI-12 yrs./11 yrs./ 23 yrs.	Construction Accounting Mgt.	Construction Administration	The J. G. White Engineering Corp.	Field Office Manager
J.R. Dufort	B.S. Business Administration	Project Budget Manager	2 yrs./37 yrs./39 yrs.	Budget & Cost Accounting	Cost Engineering	General Electric Company	Senior Cost Engineer

+Registered P.E., Washington.
✓Registered P.E., other state(s).

WASHINGTON PUBLIC POWER SUPPLY SYSTEM

MAY 1981

POWER GENERATION
K. T. PERKINS
DIRECTOR

- **HANFORD/PACKWOOD GENERATING PROJ.** — A. McDONALD, PLANT MANAGER
- **GENERATION MGMT. SERVICES** — G. E. DEEGAN, MANAGER
- **POWER GENERATION SERVICES** — B. K. McLEOD, MANAGER
- **WNP-1/4 OPERATIONS** — R. E. SMITH, ASS'T PROGRAM DIRECTOR
- **WNP-2 PLANT** — J. D. MARTIN, MANAGER
- **WNP-3/5 PLANT** — P. F. AHERN, MANAGER

8105041

KENWOOD T. PERKINS

POSITION: DIRECTOR, POWER GENERATION

EDUCATION: BA, PE

EXPERIENCE:

1971 - 1980	MANAGER, STARTUP AND OPERATIONAL SERVICES, NUCLEAR SERVICES CORPORATION • STARTUP CONSULTANT
1969 - 1971	PRESIDENT, TECHNICAL EQUITIES CORPORATION, SAN JOSE • MANAGEMENT
1955 - 1969	MANAGER IN ATOMIC POWER EQUIPMENT DEPARTMENT, GENERAL ELECTRIC, SAN JOSE • STARTUP • OPERATOR TRAINING • SIMULATOR DESIGN, CONSTRUCTION AND OPERATION
1954 - 1955	LEAD ENGINEER, KNOLLS ATOMIC POWER LABORATORY • DESIGN • STARTUP AND PREOPERATIONAL TESTING
1944 - 1954	LEAD ENGINEER, DUPONT AND GENERAL ELECTRIC, OAK RIDGE • STARTUP AND OPERATION PRODUCTION REACTOR • DESIGN AND CONSTRUCTION LIAISON
1941 - 1943	ENGINEER, METALLURGICAL LABORATORY, UNIVERISTY OF CHICAGO/DUPONT • FUEL ELEMENT DEVELOPMENT

INFORMATION ON POWER GENERATION MANAGEMENT AND STAFF

DEPARTMENT: POWER GENERATION DIRECTORATE

AS OF: August 31, 1981

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
DEEGAN, G. E.	BS-Math & Physics	Manager, Generation Mgmt. Services	1.5/29.5/31	Mgmt. of Reactor Operations/Maint.	Nuclear & Liquid Metal R&D Mgmt.	Argonne Nat'l. Lab	EBR-II Operations Mgr.
HANNAH, K. J.	BS-Welding & Metallurgical Engineering	Supv., Nondestructive Examination & Inspection	4/15/19	Nondestructive Examination; Preservice & Inservice NDE Evaluation; Registered Quality Engineer	Failure Analysis in BWR Systems	GE; Automation Industries; Lockheed Missiles & Space; Rocketdyne	Sr. Engr.; Mat'ls. & Chemistry; Research Engr.; Mfg. Engr.; Field Engr.
HOLDER, J. R.	BS-Math & Engineering	Program Manager, Power Generation	2/ 9/11	Nuclear Power Plant Operations/Startup & Testing/Management; Operational Design Review; Startup & Operational Programs Development & Management		Carolina Power & Light	Plant Supt.; Plant Startup Engr.; Sr. Engr.; Refueling Engr.; Nuclear Engr.; Sr. Reactor Engr.
HOLZENDORF, W. F.	HS; 1 Yr. College; Navy Nuclear Power School; Navy Engineering Officer School	Acting Supv., Maintenance Services (Sr. Maintenance Specialist)	1/23/24	Mechanical/Systems	Operations Engineer	Westinghouse Hanford; U.S. Navy	Operation Support Engr.; Seaman
ITTNER, J. P.	BS-Marine Engineering	Principal Engineer	6.5/9.5/16	Nuclear Operations & Startup; Training Simulator Project Management.	Reactor Operations; Nuclear Technology/Systems Trng.; Planning & Scheduling	Babcock & Wilcox; First Atomic Ship Transport, Inc	Site Operations Engr./Technical Support Supv.; Reactor Operator
KUBENKA, J. H.	BS-Occupational Education	Supv., Support Training	1/15/16	Reactor Operations/Maintenance/Training; Project Engr.	Computer Programming & Statistical Analysis	United Nuclear Corp.; US Navy	Trng. Rep.; Nuclear Power School Advisor

Registered P. E., Washington.
Registered P. E., other state(s).

INFORMATION ON POWER GENERATION MANAGEMENT AND STAFF

DEPARTMENT: POWER GENERATION DIRECTORATE

AS OF: August 31, 1981

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
McLEOD, B. K.	Navy Nuclear Reactor Program	Manager, Power Generation Services	8/12/20	Nuclear Power Plant Operations; Quality Assurance	Nuclear/Industrial Safety; Training	United Nuclear Corp.; NRC; General Electric Co.; U.S. Navy	Safety/Licensing Mgr.; Principal Reactor Insp.; Shift Mgr.; Reactor Operator
NOYES, C. R. †√√	BSME; MSNE; Registered Professional Engr. (3 states)	Manager, Power Generation Maintenance	2 Mos./24/24	Mechanical Engineer; Nuclear Engineer; Nuclear & Mechanical Maintenance; Reactor Operations; Navy Nuclear Programs	Master Precision Machinist; Special Tool Design	U.S. Navy; U of Nebraska; Nebraska Public Power Dist.; U of Missouri	Nuclear Lab Tech./Machinist; Nuclear Engr. Asst.; Engrg. Supv.
PARTRICK, R. E.	BS-Naval Science; Navy Nuclear Power School	Supv., Generation Coordination	6.5/6.5/13	Reactor Operations/ Maintenance	Maintenance Engrg. Mgmt.	Westinghouse Hanford	Maint. Dev. Engr. Sr.
PERRY, J. F.	65 Units of College-Math & Engineering	Supv., Simulator Training Systems	5/10/15	Electronics Technician; Simulator Procurement & Operations	Previously licensed as SRO on Indian Point	Con Edison of New York; U.S. Navy	Sr. Simulator Instructor; Reactor Operator
ROSENECK, J. B.	USN-ETA/ETV/NPTU; GE-STE/BWRTC	Supv., Operational Training	2.5/18.5/21	Nuclear Analyst; Reactor Plant Engineering; Reactor Core Design & Analysis; Plant Operation; Shift Test Engr.; Shift Supv.,; USN/RO/RT; Startup Engineer; Supv., Turnover	Computer Programming Electronic Maintenance	GE-NEBG & KAPL U.S. Navy	Shift Supv.; Shift Test Engr.; Reactor Control LPO

† Registered P. E., Washington

√ Registered P. E., other state(s)

INFORMATION ON POWER GENERATION MANAGEMENT AND STAFF

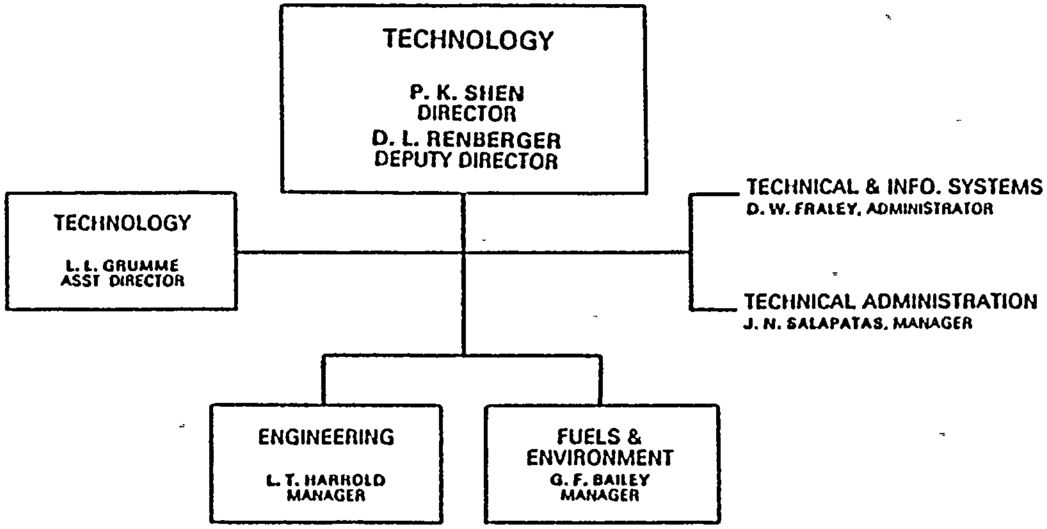
DEPARTMENT: POWER GENERATION DIRECTORATE AS OF: August 3, 1981

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
TERRASS, M. S. ✓	BS-Naval Science; MBA; Navy Advanced Nuclear Power School	Principal Engineer	3.5/17/20.5	Nuclear Engineering; PE/Nuclear; Education & Training; Nuclear Plant Operations (BWR & PWR)	Nuclear Plant Testing & Maintenance (BWR & PWR)	U.S. Navy; IBM; Nuclear Services Corp.; California Energy Commission	Naval Officer; Data Processing Marketing Rep; Specialist Engr; Energy Facility Site Planner
CHUBB, K. L.	Technical Institute Graduate; GE-Computer Maintenance	Supv., Instrument Maintenance & Calibration	3/23/26	Nuclear Plant Support	Computer Maintenance; Standards Laboratory Supervision; Instrument Technical Instruction	Westinghouse Hanford; Battelle NW; General Electric	Calibration & Computer Maintenance Mgr.; Computer Maintenance Spec.; Computer Field Engr.; Instrument Technician

Registered P. E., Washington. †
Registered P. E., other state(s). ✓

WASHINGTON PUBLIC POWER SUPPLY SYSTEM

MAY 1981



8105041

PETER K. SHEN

POSITION: DIRECTOR, TECHNICAL

EDUCATION: BS, MS, PHD

EXPERIENCE:

1975 - 1980 DEAN AND RESIDENT DIRECTOR, JOINT CENTER FOR GRADUATE STUDY, RICHLAND, WA

- RESEARCH
- TEACHING
- FUEL CYCLE DEVELOPMENT

1973 - 1975 SUPERVISING ENGINEER, SOUTHERN CALIFORNIA EDISON CO.

- NUCLEAR ANALYSIS
- FUEL MANAGEMENT
- LICENSING

1970 - 1973 ENGINEER, SOUTHERN CALIFORNIA EDISON CO.

- NUCLEAR FUEL CYCLE MANAGEMENT
- NUCLEAR ANALYSIS

INFORMATION ON TECHNOLOGY MANAGEMENT AND STAFF

DEPARTMENT: TECHNOLOGY

AS OF: 7/31/81

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
+ P. K. Shen	BS Chemical Engr. MS Chemical Engr. PhD Nuc. Engr.	Director, Technology	1 yr./14 yr./15 yr.	Reactor Analysis, Nuclear Systems Design, Nuclear Reactor Safety, Nuclear Fuel Mgmt.	Reactor Operation, Licensing, Nuclear Waste, Radiation Effect Materials	Univ. of Washington	Dean and resident Director, Professor Nuclear Engineering
D. L. Renberger	BS Nuclear Engr.	Deputy Director, Technology	10 yr./13 yr./23 yr.	Nuclear Plant Systems Performance and Testing	Nuclear Safety and Licensing	Douglas United Nuc.	Supervisor Regulatory Unit
✓ L. L. Grumme	BS Nuclear Engr.	Assistant Director, Technology	10 yr./11 yr./21 yr.	Nuclear Fuel procurement Nuclear Fuel Mgmt. Management	Reactor Operations Environmental Licensing Nuclear Safety	Douglas United Nuc.	Sr. Engr.
†✓ J. N. Salapatas	BS Industrial Engr. MS Management	Manager, Technical Admin.	1 yr./25 yr./26 yr.	Project Management Proj. Controls/ Nuclear Engr. & Construction	Management Systems, Organization Planning and Development and Training	Florida Power & Light	Manager, Project Control Services
D. W. Fraley	BS Engineering Sciences PhD Systems Engineering	Administrator, Technical and Information Systems	4 mo./12 yr./12 yr.	Systems Analysis, Decision Analysis, Operations Research Applied to Nuc. Industry	Computer Science, Information Systems, Hardware & Analysis	Battelle	Sr. Research Scientist, Nuclear Energy Systems

✓ Registered P.E., Washington.
† Registered P.E., other state(s).

INFORMATION ON ENGINEERS IN ENGINEERING DIVISION

DEPARTMENT: Engineering Division Staff (42100) AS OF: 7-31-81

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS	EXPERIENCE OTHER	EXPERIENCE TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
CL Fies	BSEngPhys, MSNE, MBA	Program Mgmt Spec	7	10	17	Program Mgmt	-----	Westinghse	Sr Engr
✓ LT Harrold	BSME, BAEcon	Engr'g Division Manager	8	8	16	Mechanical	ISI	DUN	Sr Engr

✓ Registered P.E., Washington.
† Registered P.E., other state(s).

INFORMATION ON ENGINEERS IN ENGINEERING DIVISION

DEPARTMENT: Mechanical Engineering (42500)   AS OF: 7-31-81

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS	EXPERIENCE OTHER	EXPERIENCE TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
✓ WD Bainard	BSChE	Consulting Engr	9.25	20	29.25	Corrosion	Water Quality/N-C	DUN	Sr Engr
D Burns	BSMetl, PhD	Supv, Materials & Components	3.75	5	8.75	Metallurgy	Welding	IL Rsch Inst	Rsch Mtlgst
†DM Bosi	BSME, MSME	Sr Engr	0	9	9	Engr'g Mechanics	Heat Transfer	Westinghse	Sr Engr
MK Chakravorty	PhD/Str.Dyn.	Principal Engr	2.75	7	9.75	Earthquake Engr'g	Structural Dynamics	Stone/Webstr	Prin Engr
JR Cole	BSME	Engr I	.5	9	9.5	Machine Design	-----	WSH	Sr Engr
DW Cutting	BSMetE	Principal Engr	1.75	13	14.75	Welding	Metallurgy	WGC	Chief Wldg Eng
TM Erwin	BSMetE	Associate Engr	.5	0	.5	Weld'g/Metallurgy	Welding processes	MI NavShpyd	Engr in Trng
✓†HE Good	BSME, MSCE	Supv, Engr'g Mechanics	3	17	20	Structural Dynamics	Seismic/Soil Mech.	Stone/Webstr	Lead Engr
TF Hoyle	BSME	Senior Engr	3	7	10	ISI/PSI/NDE	Welding	IL Pwr	ISI Engr
JD Husted	AAS	Engr'g Assistant	1.75	.25	2	Engr'g Documentn	-----	UNI	Lab Tech
PJ Inserra	BSEngTech	Associate Engr	1	.25	1.25	Welding	Metallurgy	DiabloCanyon	QA Inspctr
RJ Lauzon	BSME	Senior Engr	2	20	22	Welding	Piping	Consultant	Self-Employd
✓ RL Loundagin	BSME	Senior Engr	3.5	24	27.5	Piping (ASME)	Stress Analysis	UNI	Sr Engr
RA Moen	BSMetE	Senior Engr	1.5	18	19.5	Metallurgy	ASME	Westinghse	Sr Engr
†EL Morales	BSME	Senior Engr	3.75	13.5	17.25	Stress Analysis	Piping	Stearns-Rgrs	Engr III
✓ JC Mowery	BSME	Principal Engr	8.75	12	20.75	Piping	Systems Engr'g	HUICO	Proj Engr
DA Murdock	BSME	Engr I	.5	10.5	11	NDT (Ultrasonics)	-----	NucEngySvcs	NDE Engr

✓ Registered P.E., Washington.
† Registered P.E., other state(s).

INFORMATION ON ENGINEERS IN ENGINEERING DIVISION

DEPARTMENT: Mechanical Engr'g - CONTINUED (42500)

AS OF: 7-31-81

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL			PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
✓ JE Parker	BSME	Senior Engr	5.5	6	11.5	Rotating Equipt	Systems Engr'g	GE	Serv Engr
✓†DW Porter	BSNE	Manager, Mechanical Engr'g	4.75	3.5	8.25	ISI/PSI/NDE	Systems Engr'g	Bechtel	Sr Nuc Engr
LR Prill	BSME	Engr I	.25	9	9.25	Ops/Maint/Design	Personnel Trng	US Navy	Sub Officer
DP Ramey	BSChE	Engr I	.75	5	5.75	ISI/PSI	-----	Rockwell	Plt Engr
†MP Reis	BAMath, MSNE	Engr I	1.25	4	5.25	Systems Engr'g	ISI/PSI/NDE	PG&E	T&SU Engr
CD Scott	BSME	Engr II	1.25	3	4.25	Piping Systems	Heat Transfer	WSH	Engr
K Singh	BSME	Senior Engr	4.25	8.5	12.75	ISI/PSI/NDE	Piping	Bechtel	Pipg Engr

✓ Registered P.E., Washington.
† Registered P.E., other state(s).

INFORMATION ON ENGINEERS IN ENGINEERING DIVISION

DEPARTMENT: Civil Engineering & Services (42600)

AS OF: 7-31-81

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS	OTHER	TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
PJ Adair	None	Sr Engr'g Records Analyst	6.25	0	6.25	Documentation	-----	-----	-----
AR Bright	BCS	Sr Admin Engr	4.75	15	19.75	Engr'g Admin	Accounting	Booing	QA Supv
KM Brun	BASecEduc	Engr'g Records Analyst II	1.5	9.25	10.75	Equipt Qualif	Documentation	CPCO	Plt Clerk
BM Burnside	None	Engr'g Records Analyst I	1.25	4	5.25	Documentation	Comp Applic	B&R	Technician
PE Campbell	ASNucTech	Engr'g Records Analyst II	0	3	3	-----	-----	Westinghse	Technician
KN Chick	BSAnthrp	Engr'g Records Analyst I	3.5	3	6.5	Equipt Qualif	Comp Applic	-----	-----
MA Coffman	AA	Engr'g Records Analyst II	0	4	4	Documentation	-----	Westinghse	Technician
CR DePoe	None	Senior Engr	3.25	21.5	24.75	Mechanical	Electrical	SD St Univ	Manager
VK Duggal	BAArcht	Senior Designer	2	11	13	Design/Drafting	-----	B&R	Designer
✓†WC Flynn	BSCE	Senior Engr	2.75	21	23.75	Fac Des/Install	Struct Design	Pt/Olympia	Chief Engr
FV Gomulka	BA Hum.Resc	Engr'g Records Analyst I	1.25	1.5	2.75	Documentation	Comp Applic	Westinghse	Data Clerk
WS Hardy	None	Engr'g Records Analyst II	2.25	3	5.25	Documentation	-----	Bechtel	Elect Clerk
NH Kempt	BSEE	Senior Engr	3.25	18	21.25	Electrical	-----	Vitro	Sr Engr
WH Knolle	None	Engr I	3.5	37	40.5	Elect Eng/Design	Cost Estimating	Vitro	Sr Estmtr
LE Maravilla	None	Admin Specialist	7.75	4	11.75	Administration	Secretarial	Juv Court	Sec'y
KF McAndrew	AAS Tech Illust	Engr'g Asst (Drafting)	3.75	7	10.75	Design/Drafting	-----	Bovee/Crail	Drafter

✓ Registered P.E., Washington.
† Registered P.E., other state(s).

INFORMATION ON ENGINEERS IN ENGINEERING DIVISION

DEPARTMENT: Civil Engr'g & Svcs - CONTINUED
(42600)

AS OF: 7-31-81

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS	OTHER	TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
BM Nichols	None	Sr Engr'g Records Analyst	5.5	17	22.5	Documentation	Engr'g Admin	City/Rchld	Sec'y
EV Norris	BAArcht	Senior Engr	1.25	30	31.25	Architectural	-----	Rockwell	Proj Engr
CO Romine	BSEE	Supv, Engr'g Services	4	10	14	Engr'g Admin	I&C	Miles Labs	Mgr
✓†ER Rybarski	BSCE	Principal Engr	8.75	25	33.75	Civil	-----	Westinghse	Sr Engr
JR Smith	None	Senior Designer	1.25	14	15.25	Design/Drafting	-----	Rockwell	Designer
✓ HW Stivers	BSCE	Supv, Civil & Facs Engr'g	8.25	27	35.25	Civil	Structural	Westinghse	Sr Engr
✓†WK Stockdale	BSCE,MSCE,PhD	Manager, Civil Engr'g & Svcs	3.5	27	30.5	Struct. Dynamics	Civil/Soil Dynamcs	US Army	Colonel
†HE Wellsfry	BSCE	Senior Engr	3.5	15	18.5	Civil	Structural	Cty/Fremont	Engr
CL Whitcomb	BSEduc	Sr Engr'g Records Analyst	5.5	4	9.5	Documentation	Engr'g Admin	-----	-----

✓ Registered P.E., Washington.
† Registered P.E., other state(s).

INFORMATION ON ENGINEERS IN ENGINEERING DIVISION

DEPARTMENT: Electrical/I&C Engineering (42700)

AS OF: 7-31-81

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS	EXPERIENCE OTHER	EXPERIENCE TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
RL Abbott	BSEE	Senior Engr	2.25	12	14.25	Elect Controls	Equipt Qualif	B&W	Sr Engr
RD Bass	BSEE	Engr I	.5	6	6.5	I&C Engr	-----	Westinghse	Adv Engr
† M Basu	BSEE, MSEE	Senior Engr	1.25	6	7.25	Systems Engr'g	Equipt Qualif	Ebasco	Engr
†SR Basu	BSEE, MSEE	Supv, Electrical Engr'g	2.25	15	17.25	Elect Systems	Licensing	Bechtel/B&R	Supv
EF Boler	AAElectron	Engr II	1.5	10	11.5	Simulators	Electron Hardware	UW	Inst Tech
MR Compton	BSPhys	Senior Engr	5.25	9	14.25	Computers	Simulators	Battelle	Dvlpt Engr
✓ AJ Ebens	BSME, MSME	Program Mgmt Spec	6.75	10.25	17	Program Mgmt	Contract Admin	Westinghse	Proj Engr
WE Faraone	BSEE	Engr I	4.75	6	10.75	Equipt Qualif	Elect Field Engr'g	Bechtel	Fld Engr
†RE Green	None	Senior Engr	5.25	11	16.25	I&C Engr'g	Systems Engr'g	GE	Engr
FR Guyer	BSEE	Consulting Engr	7	18	25	Simulators	Electrical	Westinghse	Sr Engr
†AN Joshi	BSEE	Senior Engr	2.25	20	22.25	I&C Systems	Equipt Qualif	Bechtel	Sr Engr
JD Lodge	BSEE, MSNE	Senior Engr	1.5	10	11.5	Simulators	Electrical	Westinghse	Sim Engr
JT Person	BSEE	Engr II	0	2	2	Elect Distrb.	-----	EG&G	Engr
✓ JE Rhoads	BSEE	Supv, Equipt Qualif Engr'g	5	3.5	8.5	Equipt Qualif	Electrical	Bechtel	Fld Engr
RC Ruckdeschel	BSEE	Senior Engr	0	18	18	Systems	-----	Westinghse	Sr Engr
† U Shah	BSEE, MSInstE	Manager, Elect/I&C Engr'g	4.25	13	17.25	I&C	Sfty/Reliblty	NucSvc/3M	Sr Const.

✓ Registered P.E., Washington.
† Registered P.E., other state(s).

INFORMATION ON ENGINEERS IN ENGINEERING DIVISION

DEPARTMENT: Elect/I&C Engr'g - CONTINUED (42700)

AS OF: 7-31-81

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS	OTHER	TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
JP Starr	None	Engr I	5.75	20	25.75	Simulators	Electrical	Westinghse	Engr
JL Sullivan	BSEE	Senior Engr	1.25	20	21.25	Equipt Qualif	Reliability Engr'g	Hoffman	Lead Engr
✓ DT Thonn	BSEE	Principal Engr	8.5	22	30.5	Elect Systems	Protective Relay	ARCHO/GE	Spec
CJ Zeamer	BSNETech	Engr I	.5	20	20.5	Equipt Qualif	Maintenance/Elect	B&R	Engr

✓ Registered P.E., Washington.
† Registered P.E., other state(s).

INFORMATION ON ENGINEERS IN ENGINEERING DIVISION

DEPARTMENT: Nuclear Engineering (42800) AS OF: 7-31-81

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS	OTHER	TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
†SM Berry	BSME	Engr I	1.25	10	11.25	Systems Engr'g	Fuel Handling	Rockwell	Sr Engr
DE Bush	BSChE,MSNE,PhD	Principal Engr	4.75	12	16.75	Thermal Hydraulics	Containment Perf.	Stone/Wbstr	Sr Engr
†LC Chan	BSME	Engr I	0	7.5	7.5	Mechanical Systems	ThermHydr Analysis	CT Main	Mech Engr
†KR Conn	BSAE, MSAE	Engr I	4.25	19.5	23.75	Systems Analysis	Heat Balance	Gilbert Asc	Sr Engr
†GL Gelhaus	BSEPhys, MSNE	Manager, Nuclear Engr'g	7.25	7	14.25	Safety Analysis	Fuel Mgmt	TVA/BNW	Nuclear Engr
RL Heid	BSME	Principal Engr	7.5	13	20.5	Syst Perf Analysis	Component Design	Westinghse	Sr Engr
TW Horning	BSBE, MSNE	Engr I	0	5	5	Program Mgmt	Sfty/Environ Mntrg	Westinghse	Mgr,CS Anly
TH Keheley	BSME	Engr I	0	10	10	Heat Transfer	-----	Ecodyne	R&D Engr
WH Kelso	BSME	Program Mgmt Spec	3.75	17	20.75	Fuel Handling	-----	Rockwell	Sr Engr
DB Kreig	BSME	Associate Engr	0	2	2	-----	-----	NM Rsch Fac	Rsch Asst
KS Lund	BSNE	Associate Engr	.25	.75	1	-----	-----	PS NavShpyd	Eng Tech
†FJ Markowski	BSME, MSME	Senior Engr	1.75	15	16.75	Reliability	Plt Transt Analys	Westinghse	Sr Engr
TB McCall	BSEE,MSNE, PhD	Program Mgmt Spec	.25	18	18.25	Plt Analy/Design	-----	Westinghse	Prin Engr
DM Myers	BSEngPhy, MSNE	Supv, Systems & Reliablty	0	10	10	Syst Analy/Design	-----	Rogers/Asc	Sr Mgr
BD Ngo	BSME	Associate Engr	.25	0	.25	Mechanical	-----	-----	-----
RJ Nicklas	BSChE, MBA	Supv, Nuclear/Chem Engr'g	7.75	12.5	20.25	Nuclear/Chemical	Systems Engr'g	WI-Mich Pwr	Engr

✓ Registered P.E., Washington.
† Registered P.E., other state(s).

INFORMATION ON ENGINEERS IN ENGINEERING DIVISION

DEPARTMENT: Nuclear Engr'g - CONTINUED (42800) AS OF: 7-31-81

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS	OTHER	TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
FE Owen	BSChE	Senior Engr	8.25	25	33.25	Radiation Protect.	-----	UNI	Sr HP
WW Roberts	BSChE	Engr I	5.25	8	13.25	Corrosion	-----	ARCHO	Sr Engr
MR Steelman	BSEcon, BSAeroE	Engr I	3.5	2	5.5	Computer Engr'g	-----	-----	-----
RT Steen	BSME	Engr I	3.5	5	8.5	Radwaste	Systems Engr'g	Cmwlth Edisn	Gen Engr
JP Thorpe	BSNE	Engr I	1	4	5	Shielding	Fuels Cycles	NN Shpbldg	Engr
WW Waddel	BSEPhys, MSNE	Supv, Tech Programs	7	13	20	Program Mgmt	-----		

✓ Registered P.E., Washington.
† Registered P.E., other state(s).

INFORMATION ON ENGINEERS IN TECHNICAL DIVISION

DEPARTMENT: FUELS & ENVIRONMENT DIVISION

AS OF: AUGUST 5, 1981

Name	Degree(s)	WPPSS Position Title	Experience WPPSS	Experience Other	Experience Total	Principal Area of Expertise	Secondary Area of Expertise	Previous Employer	Title
GF Bailey (✓)	BS Chemical Engineering BS Business MS Nuclear Engineering	Manager, Fuels & Environment Division	7	16	23	Nuclear Engineering/ Management	Energy Alternatives/Environmental Engineering	Westinghouse Hanford	Manager, Environmental Engineering
RG Spencer	--	Technical Division Administrative Specialist	7	26½	33½	Financial	Administrative	UNC	Manager, Cost and Property

✓ Registered PE in Washington

Name	Degree(s)	WPPSS Position Title	WPPSS	Other	Total	Expertise	of Expertise	Employer	Title
JR Worden	BS/MS Physics	Manager, Fuel Supply	7	17	24	Fuel Management	Nuclear Analysis	West/BNW/GE	Fuel Project Engineer
BK Boyd	BS Met Engr MS Engr Mgt	Supervisor, Fuel Services	4½	17	21½	Fuel Contract Admin. (Tech)	Fuel Quality Assurance	Argonne/ Texas Instruments/ GE	Procurement Engineer
LS Pace	BS Mathematics	Fuel Cost Engineer	5	--	5	Fuel Cost	Project Planning and Scheduling	--	--
JE Sammis	BS Ceramic Engineer	Fuel Project Engineer	7	6	13	Fuel Contract Admin (Tech)	Fuel Development	West.Hanf./ BNW	Fuel and Core Component Procurement Engineer
WG Jolly	BA Math MA Econ	Economic Evaluations Engineer	½	5	5½	Economic Analysis	Budget Forecast-	West.Hanf.	Economic Studies Engineer
JR Young (✓,†)	BS CE SM CE MBA	Principal Special Studies Engineer	4	29	33	Engineer Studies/ Management	Cost Benefit Analysis/Environmental Assessment/ Economic Analysis	BNW/DUN/GE	Staff Engineer

✓ Registered PE in Washington
† Registered PE in other state(s)

INFORMATION ON ENGINEERS IN TECHNICAL DIVISION

DEPARTMENT: ENVIRONMENTAL PROGRAMS DEPARTMENT

AS OF: AUGUST 5, 1981

Name	Degree(s)	WPPSS Position Title	Experience WPPSS	Experience Other	Experience Total	Principal Area of Expertise	Secondary Area of Expertise	Previous Employer	Title
KA McGinnis	BS Ag Econ MS Ag Econ	Economics Evaluations Engineer	1½	4	5	Economics	Engineer Economics/Socioeconomics	BNW	Economist
KR Wise (✓)	BS Mechanical Engineering MS Mechanical Engineering	Supervisor, Environmental Engineering	8	7	15	Supervision	Environmental Engineering	Westinghouse Hanford	Development Engineer
W Davis, III	BS Biology MS Ecology MS Management Science	Supervisor, Environmental Science	2	11	13	Ecology	Statistics	Yankee Atomic Electric Company	Environmental Scientist
RA Chitwood	BA Physics Chemistry Mathematics	Manager, Environmental Programs	10	15	25	Environmental/ Regulatory Management	Siting/Licensing Management	DUN	Senior Engineer
JP Chasse (†)	BS CE MS CE-Water Resources Engineering	Senior Environmental Engineer	4	7+	11	Environmental Engineering		US EPA	Civil/Environmental Engineer
AC Rutz	BS Biology MPH-Water Quality Environmental Chemistry	Environmental Engineer	2	8	10	Environmental Engineering	Water Quality/ Environemntal Chemistry	Klamath County, OR	Director of Environmental Health

✓ Registered PE in Washington
† Registered PE other state(s)

INFORMATION ON ENGINEERS IN TECHNICAL DIVISION

DEPARTMENT: ENVIRONMENTAL PROGRAMS DEPARTMENT

AS OF: AUGUST 5, 1981

Name	Degree(s)	WPPSS Position Title	Experience WPPSS	Experience Other	Experience Total	Principal Area of Expertise	Secondary Area of Expertise	Previous Employer	Title
MK Thompson	BS Geology Grad Study in Hydrology	Environmental Engineer	1½	3 res. 4 ind. & gov't	8½	Hydrology Geology	Permitting and Regulatory Compliance	Utah Dept. of Natural Resources	Engineering Geologist/ Senior Hydrologist
C Van Hoff	--	Senior State Liaison Specialist	4	5	9	Regulatory Interface	Legislative Reviewer	State of Idaho	Legislative Assistant
JC Kutt	BS Chemistry	Environmental Scientist	3 mo	10	10	Chemistry	Environmental Scientist	Battele NW Laboratory	Scientist
GS Jeane	Minor Chemistry BS Fisheries	Environmental Engineer - Biologist	5	6½	11½	Mitigation Fisheries Pollution	Water Quality	Washington State Department of Ecology	Environmentalist
WA Kiel	BS Geology BS Oceanography	Geologist	5½			Environmental Geology			
AM Lee	MUP Urban Planning; BA Urban and Regional Planning	Socioeconomic Coordinator	3	1	4	Socioeconomic Monitoring Mitigation	Land Use (physical planning)	Washington State Office of Community Development	Administrative Intern
ML Miller	BS Physical Science MS Health Physics	Environmental Scientist	1½	3	4	Health Physics	Computer Programmer	Exxon	

INFORMATION ON ENGINEERS IN TECHNICAL DIVISION

DEPARTMENT: ENVIRONMENTAL PROGRAMS DEPARTMENT

AS OF: AUGUST 5, 1981

Name	Degree(s)	WPPSS Position Title	Experience: WPPSS	Experience: Other	Experience: Total	Principal Area of Expertise	Secondary Area of Expertise	Previous Employer	Title
JE Mudge	PhD MS MEd BS	Senior Environmental Scientist	3	8	11	Aquatic Ecology.	Physiology Bioassay	Metropolitan Edison Company/ Texas Instrument Inc.	Supervisor, Radiation Safety and Environmental Engineer/Aquatic Ecology Task Leader
FD Quinn	BS Meteorology 2+ Yrs Grad School	Senior Environmental Scientist	1	20	21	Air Pollution Meteorology	Scientific Systems Analyst	USAF/Air Wea. Svc.	Technical Division Meteorological Systems Analyst
LS Schleder	AA Zoology, Environmental Studies	Environmental Technician	2 (8mos Security Guard for WPPSS)	11	13	Environmental Technician	Security Guard	Industrial Fiberglass/ Shelbies Fiberglass	Miscellaneous Fabrication

INFORMATION ON ENGINEERS IN ENGINEERING DIVISION

DEPARTMENT: WNP-2 ENGINEERING 62900

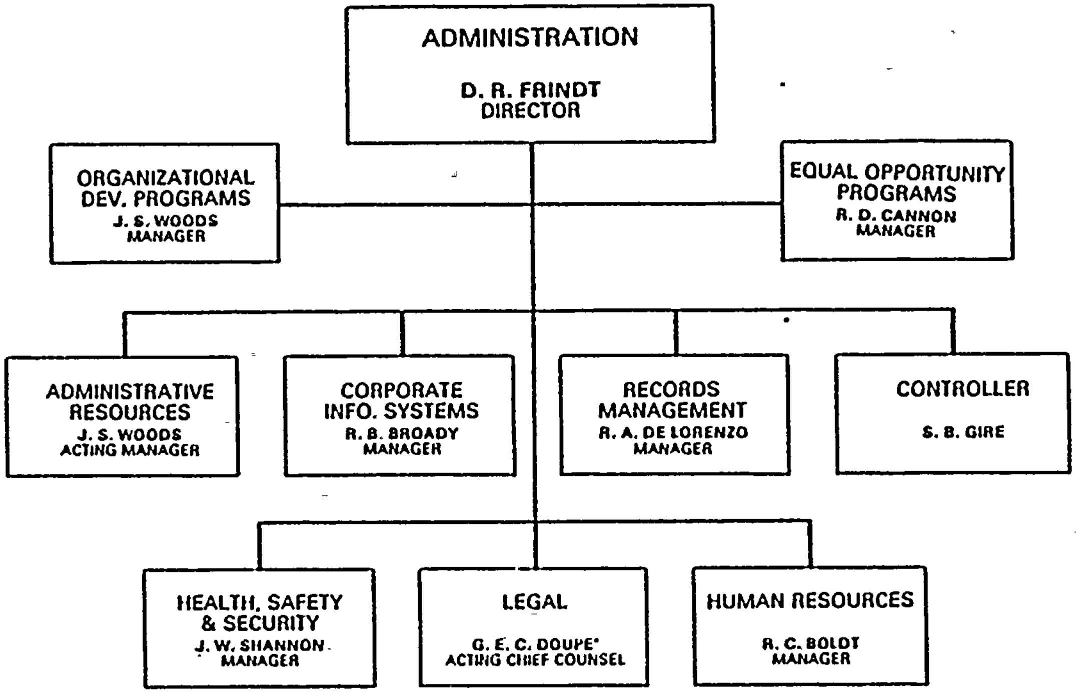
AS OF: 8-6-81

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS	OTHER	TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
HL Bennett	BSME	Senior Engineer	5.25	21	37	HVAC	Valves	Industrial Pipe/Supply	Equipt Engr
+✓HA Crisp	BSME, MSPE, MSCE	Project Engineering Management Specialist	1	25	26	Civil/Structural/ Management	Systems	U. S. Navy	Captain/Navy
+✓EA Fredenburg	BSCE, MBA	Principal Engineer	5.25	6	11.25	Structural	Cont.Des/Prog Mgmt	Bechtel	Sr. Engr.
PW Harness	BSME	Manager, Field Engineering (Acting)	8.75	5	13.75	Systems Engrg.	Rotating Equipm't.	UNC	Plant Engineer
BA Holmberg	BS Marine Engr.	Deputy Project Manager-Engrg.	2.75	20	22.75	Management/Nuclear/ Operations	Mechanical	U.S. Navy	Commander/Navy
+ JH Ralphs	BS Engineering MS Gen. Engrg.	Senior Engineer	3.5	25	28.5	Mechanical	Chemical	Bechtel	Sr. Field Engr
TA Stanley	BSME	Project Engrg. Mgmt Spec.	4.25	8.5	12.75	Systems	Construction Engrg.	ITT	Project Engr.
DC Timmins	BS Physics	Project Engrg Mgmt Spec	5.3	7.7	13	Nuclear/Mech.	Chemical Engrg/ Shielding	Bechtel	Supervisor Nuclear Analys
✓ LW Vance	BSCE	Project Engrg Mgmt Spec	6	10	16	Civil	Highway	Wash. State Dept.Transportation	Sr. Engineer
✓ MF Wiitala	BSEE	Principal Engineer	9	25	34	Electrical	---	Westinghouse	Sr. Engineer

✓Registered P. E., Washington
+ Registered P. E., Other State(s)

WASHINGTON PUBLIC POWER SUPPLY SYSTEM

MAY 1981



NOTE THE CHIEF COUNSEL AND THE CONTROLLER, BOTH OF WHOM REPORT TO THE DIRECTOR, ADMINISTRATION, REPORT DIRECTLY TO THE MANAGING DIRECTOR IN AREAS INVOLVING ATTORNEY/CLIENT OR FISCAL/CLIENT RELATIONSHIP WITH THE MANAGING DIRECTOR AND/OR BOARD OF DIRECTORS

8105041

DWIGHT R. FRINDT

POSITION: DIRECTOR, ADMINISTRATION

EDUCATION: BA, MBA

EXPERIENCE:

1979 - 1981 BUSINESS CONSULTANT

- MANAGEMENT

1977 - 1979 CHIEF OPERATING OFFICER, STANDARD EQUIPMENT INC.

- MANAGEMENT
- LOGISTICS
- CONTRACT ADMINISTRATION
- FINANCING

1971 - 1977 VICE PRESIDENT, STANDARD EQUIPMENT INC.

- HEAVY CONSTRUCTION FIELD MANAGEMENT

1970 - 1971 OPERATIONS MANAGER, MULLEN CORPORATION

- CONSTRUCTION MANAGEMENT

1968 - 1970 GENERAL MANAGER, MULLEN MINING DIVISION

- MINING PROJECT DIRECTION

WASHINGTON PUBLIC POWER SUPPLY SYSTEM

AUGUST 1981

- **HEALTH, SAFETY & SECURITY** — J. W. SHANNON, MANAGER
  - **IND. SAFETY & FIRE PROTECTION** — W. E. TAYLOR, MANAGER
    - LOSS CONTROL SERVICES — R. M. GOUGH
    - PROJECT SAFETY — W. E. TAYLOR, ACTING
  - **SECURITY** — S. R. TELANDER, MANAGER
    - ADMINISTRATIVE SECURITY — J. W. KLINGELHOEFER, SUPERVISOR
    - SECURITY FORCE — S. H. MARTIN, CHIEF
    - SECURITY TRAINING & EVALUATION — R. J. GIVIN, SUPERVISOR
  - **RADIOLOGICAL PROGRAMS** — D. E. LARSON, MANAGER
    - RADIOLOGICAL SERVICES — R. W. CRAIG, SUPERVISOR
    - RADIOLOGICAL ASSMT. & AUDIT — R. F. HAIGHT, SUPERVISOR
    - EMERGENCY PREPAREDNESS — J. V. EVERETT, SUPERVISOR
  - **WNP-1/4 HEALTH, PHYSICS/CHEMISTRY** — V. E. SHOCKLEY, MANAGER
    - HEALTH PHYSICS — J. D. MILLS, SUPERVISOR
    - CHEMISTRY — T. M. BRUN, SUPERVISOR
  - **WNP-2 HEALTH, PHYSICS/CHEMISTRY** — R. G. GRAYBEAL, MANAGER
    - HEALTH PHYSICS — L. G. BERRY, SUPERVISOR
    - CHEMISTRY — L. D. MORRISON, SUPERVISOR
  - **WNP-3/5 HEALTH, PHYSICS/CHEMISTRY** — (OPEN), MANAGER
    - HEALTH PHYSICS — (OPEN), SUPERVISOR
    - CHEMISTRY — (OPEN), SUPERVISOR

8109042

JOHN (JACK) W. SHANNON

POSITION: MANAGER, HEALTH, SAFETY & SECURITY
EDUCATION: BS HEALTH PHYSICS (EQUIVALENCY)
EXPERIENCE:

1974 - 1981 MANAGER, HEALTH, SAFETY & SECURITY, WASHINGTON PUBLIC POWER SUPPLY SYSTEM

- ENVIRONMENTAL
- EMERGENCY PREPAREDNESS
- CHEMISTRY/RADIOCHEMISTRY
- RADIATION PROTECTION
- INDUSTRIAL SAFETY & FIRE PROTECTION
- SECURITY

1964 - 1974 HEALTH & SAFETY SUPERVISOR, DAIRYLAND POWER COOPERATIVE

- PLANT FIRE PROTECTION
- TRAINING
- INDUSTRIAL SAFETY
- CONSTRUCTION
- LACBWR HEALTH PHYSICS
- REPORT WRITING

1961 - 1964 TECHNICIAN - IRRADIATION & COOLANT TESTING, GENERAL ELECTRIC COMPANY

- COMPONENT TESTING
- REACTOR OPERATION & MAINTENANCE
- SODIUM TECHNOLOGY

1959 - 1960 SENIOR HEALTH PHYSICS ANALYST, ATOMICS INTERNATIONAL

- RADIATION & CONTAMINATION CONTROL
- WASTE DISPOSAL

1956 - 1960 STATION AGENT, WEST COAST AIRLINES

1951 - 1956 CHEMICAL OPERATOR, GENERAL ELECTRIC COMPANY

- CHEMICAL SEPARATION
- RADIOACTIVE WASTE
- STARTUP & OPERATIONS

INFORMATION ON HEALTH, SAFETY & SECURITY MANAGEMENT AND STAFF

DEPARTMENT: HEALTH, SAFETY & SECURITY

AS OF: August 1981

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
J.W. Shannon	BS Physics (equivalency)	Manager, Health, Safety & Security	7 yr./23 yr./30 yr.	Construction Startup Operations	Chemical Processing Operational Testing Radiation Protection Emergency Preparedness	Dairyland Power Cooperative	Supervisor, Health & Safety Dept.

+ Registered P. E., Washington.
✓ Registered P. E., other state(s).

INFORMATION ON HEALTH, SAFETY & SECURITY MANAGEMENT AND STAFF

DEPARTMENT: INDUSTRIAL SAFETY & FIRE PROTECTION

AS OF: August 1981

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
W.E. Taylor	--	Manager, Industrial Safety & Fire Protection	1/21/22 yrs.	Fire & Radiation Protection Health & Safety Programs Fire & Radiation Testing	Laboratory Operations Emergency Preparedness Reactor Testing	Westinghouse Hanford	Manager, Health & Safety Group
R.M. Gough	AD Fire Protection BS Trade & Industrial	Supervisor, Loss Control Services	7/13/20 yrs.	Fire Protection/ Industrial Safety	Firefighting Nuclear Operations, Research, & Construction	Westinghouse	Industrial Safety & Fire Prot. Specialist
J.C. Bell	BS Mathematics	Supervisor, WNP-2 Safety	12 yrs.	Nuclear Operations Construction, & Research	Turbine Manufacturing Utility Market Planning	Westinghouse Electric	Generation Engineer
M.J. Hoylman	--	Specialist I	4 mo./11/11 yrs.	Occupational Health Testing Emergency Medicine Industrial Safety	Power Plant Construction Personnel Labor Relations	Kaiser Engineers	Safety Supervisor
D.T. Evans	BS Mechanical Engineering	Senior Engineer-Fire Protection	2 mo./9/9 yrs.	Design	Fire Protection Systems Evaluation	Western Div. Naval Facilities Engineering	Assistant Branch Head
G.E. Towne	US Air Force	Senior Specialist	3/29/32 yrs.	Developed & implemented Hanford Fire Training Manual, Supply System Fire Brigade Training Programs & Manuals	Developed & implemented Hanford Emergency Reserve Procedures, Radiation Casualty Handling for DSHS	Rockwell Hanford Co.	Manager, Fire Equip. Services

+Registered P. E., Washington...

√Registered P. E., other state(s).

INFORMATION ON HEALTH, SAFETY & SECURITY MANAGEMENT AND STAFF

DEPARTMENT: INDUSTRIAL SAFETY & FIRE PROTECTION

AS OF: August 1981

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
K.C. Bleiler	--	Training Specialist I	1 mo./22/22 yrs.	Emergency Medical & Fire Protection	Emergency Medical Technician Certified Instructor in First Aid	Columbia Oil Co.	Vending Supervisor
D.G. Parthree	--	Supervisor, WNP-1/4 Safety & Fire Protection	1 mo./19/19 yrs.	Nuclear Construction, Startup & Operation	Industrial Safety Fire Protection Emergency Preparedness	Gibbs & Hill/ D.U.C.I.	Manager, Loss Prevention
R.A. Fisher	--	Specialist I	1/21/22 yrs.	Industrial Fireman Nucl./Chem. Operator	Fire Detection Testing Equipment & Design for Fire Protection	Interlakes School	Custodian
D.A. Smith	BA Psychology	Supervisor, WNP-3/5 Safety	3/5/8 yrs.	Power Plant Construction Safety	Fossile Fuel Construction Safety	WSH/Boecon/ GERI, HNP-2	Safety Mgr.

+ Registered P. E., Washington.
✓ Registered P..E., other state(s).

INFORMATION ON HEALTH, SAFETY & SECURITY MANAGEMENT AND STAFF

DEPARTMENT: WNP-2 HEALTH PHYSICS/CHEMISTRY

AS OF: August 1981

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
R.G. Graybeal	BS Education	Manager, Health Physics/ Chemistry, WNP-2	6/21/27 yrs.	Construction, Startup & Operations	Industrial Safety Programs	Iowa Electric Light & Power	Radiation Protection Engineer
L.D. Morrison	BS Biochemistry	Chemistry Supervisor	2/2/4 yrs.	Laboratory Analysis	Plant Chemistry Control	US Testing Co.	Research Scientist
L.G. Berry	--	Health Physics Supv.	2 mo./12/12 yrs.	Operation & Shut-down Activity Surveys	Plant Radiation Surveillance Health Physics Program	Lawrence Livermore Lab.	Health & Safety Technician

+Registered P. E., Washington.
✓Registered P. E., other state(s).

INFORMATION ON HEALTH, SAFETY & SECURITY MANAGEMENT AND STAFF

DEPARTMENT: WNP-1/4 HEALTH PHYSICS/CHEMISTRY

AS OF: August 1981

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
V.E. Shockley	BS Geology	Manager, Health Physics/ Chemistry, WNP-1/4	2 yrs./20/22 yrs.	Radiological & Safety Hazards Analysis	Respiratory & Contamination Monitoring	Consumers Power Co.	Plant Health Physicist
J.D. Mills	--	Health Physics Supv.	1/22/23 yrs.	Fuel, Air & Radiation Analysis Nuclear Rockets Testing	Radiation & Contamination Surveys	Consumers Power Co.	Supv., Radiation Protection
R.M. Brun	US Navy Nuclear Training School	Chemistry Supv.	2/16/18 yrs.	Mechanical Operator Engineering Lab. Technician	Chemistry & Radiochemistry Analysis Plant Policy Administration	Consumers Power Co.	Supv., Chemical & Radiation Protection

+Registered P.E., Washington.
✓Registered P.E., other state(s).

INFORMATION ON HEALTH, SAFETY & SECURITY MANAGEMENT AND STAFF

DEPARTMENT: SECURITY

AS OF: August 1981

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
S.R. Telander	USAF Nuclear Weapons, Security Police Officer Academy, Missile Staff Officer	Manager, Security Dept.	4/20/24 yrs.	Operational Physical Security Management	--	USAF	Nuc. Security & Law Enforcement Officer
S.H. Martin	BS Business Mgmt.	Security Force Chief	3/26/29 yrs.	Operational Physical Security Management	--	USAF	Deputy Chief of Security Police
D.H. Vorheis	AA Business Mgmt.	Security Force Captain	3/23/26 yrs.	Operational Physical Security Management	Systems & Equipment Specifications	USAF	Chief, Security Police
R.J. Marzano	BA Police Science	Security Force Captain	3/24/27 yrs.	Operational Physical Management	Security Force Technical Training	USAF	Colonel, Security Police
J.W. Klingelhoefer, Jr.	BS Engineering	Administrative Security Supervisor	2/7/9 yrs.	Nuclear Security	Army Artillery	NUSAC, Inc.	Sr. Technical Associate
R.J. Givin	BS Law Enforcement MS Criminal Justice	Supv., Security Training & Evaluation	3/6/9 yrs.	Training Systems Engineer, Installation Physical Security Officer	Security Force Training	US Military Police	Curriculum Officer

+ Registered P.E., Washington.
✓ Registered P.E., other state(s).

INFORMATION ON HEALTH, SAFETY & SECURITY MANAGEMENT AND STAFF

DEPARTMENT: RADIOLOGICAL PROGRAMS

AS OF: August 1981

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
D.E. Larson	US Navy Nuclear Power School College equivalency	Manager, Radiological Programs	7/13/20 yrs.	Health Physics Radiological Protection	Engineering Lab. Technician	LaSalle Boiling Water Reactor	Sr. Health Physics Technician
J.V. Everett	BS Computer Science/Physics MS Nuclear Engineering/Health Physics	Supv., Emergency Preparedness	3/2/5 yrs.	Emergency Planning	Radiological Engineer	Vallecitos Nuclear Center	Radiological Engineer
J.R. Allen	BS Physics ME Nuclear Engineering/Health Physics	Emergency Planning Specialist	1/3/4 yrs.	Emergency Plan & Procedure preparation	Radiation Specialist	USNRC	Radiation Specialist
D.H. Oatley	BS Biology MS Health Physics	Emergency Planning Specialist II	1/4/5 yrs.	Health Physics	Emergency Preparedness Planning Radiological Engineering Design	Portland General Electric	Health Physics Engineer
A.F. Klauss	--	Emergency Planning Specialist	3/9/12 yrs.	Law Enforcement	Emergency Planning	PA State Police	Trooper
D.H. Mannion	BS Business Administration MS Management	Emergency Planning Specialist	4 mo./10/10 yrs.	Nuclear Weapons Programs	Nuclear Weapons Operations	US Army	Executive Officer
R.F. Haight	BS Physics MS Physics	Supv., Dose Assessment & Audit Section	3/7/10 yrs.	Health Physics	Pathway Modeling, Dose Assessment	UNC Nuclear	Sr. Health Physicist

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✓Registered P. E., other state(s).

INFORMATION ON HEALTH, SAFETY & SECURITY MANAGEMENT AND STAFF

DEPARTMENT: RADIOLOGICAL PROGRAMS

AS OF: August 1981

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
G.V. Oldfield	BA Physics MPH	Health Physicist	2/7/9 yrs.	Emergency Response & Preparedness	ALARA Design Review	UNC Nuclear Industries	Sr. Health Physicist
D.B. Ottley	BS Physics	Health Physicist	2/4/6 yrs.	Radiation Control	Risk Assessment Emergency Preparedness	UNC Nuclear	Safety Engineer
J.O. Parry	BS Applied Physics	Health Physicist	1/6/7 yrs.	Health Physics	ALARA Review	Commonwealth Edison	Supv. Rad. Protection Chemistry
R.W. Craig	College equivalency Health Physics	Supv., Radiological Programs	4/18/22 yrs.	Construction Startup Operations	Critical Facility/ Fuel Fabrication	Dairyland Power Cooperative	H/P Technician
C.A. Graybeal	AA	Dosimetry Analyst	1/2/3 yrs.	Recording Radiation Exposure	Respiratory Protection Records	Richland School District	Secretary
J.D. Artis III	AA Envir. HP/C BS Envir. Science Grad Studies	Dosimetry Analyst	1/6/7 yrs.	Chemical Testing of Water & Waste Water	External Dosimetry Measurement Program	USAF	Sergeant
T.J. Froelich	BS Physics MS Envir. Health Science	Health Physicist I	1/7/8 yrs.	Health Physics	Power Plant Construction	Battelle	Research Scientist
Y.E. Derrer	--	Training Spec. I	5 mo./11/11 yrs.	Health Physics	Radiological Medical Fields	Public Service Co. of Colorado	Health Physics Technician
R.C. McGillic	US Navy Nuclear Power School	Training Spec. I	1½/10/11½ yrs.	Engineer Officer of the Watch	Training, Health Physics	Health Physics Systems, Inc.	Sr. Health Physicist

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INFORMATION ON HEALTH, SAFETY & SECURITY MANAGEMENT AND STAFF

DEPARTMENT: RADIOLOGICAL PROGRAMS

AS OF: August 1981

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
G.D. Rhinehart, Jr.	US Navy Nuclear Power School	Training Spec. I	1/10/11 yrs.	Operations Radiation Protection	Safety Fire Protection	Omaha Public Power District	H.P. Technician
R.E. Broz	BS Physiology	Training Spec. I	1 mo./10/10 yrs.	Health Physics	Emergency Preparedness	State of WA	Health Physicist

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✓Registered P. E., other state(s).

INFORMATION ON NUCLEAR SAFETY MANAGEMENT AND STAFF

DEPARTMENT: NUCLEAR SAFETY

AS OF: 8/31/81

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
G. D. Bouchey	BS Nuclear Eng. MS Nuclear Eng. PhD	Director, Nuclear Safety	10 mo./17 yr./18 yr.	Safety & Licensing Core Engineering Design	Research Radiation Detection Health Physics Research/Teaching	U. S. Dept. of Energy (FFTF)	Director, Operational & Experimental Safety Div.
S. E. Bussman	None	Administrative Assistant	1 yr./2 yr./3 yr.	Administration	Secretarial	Peppermill, Inc.	Office Manager

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✓ Registered P. E., other state(s).

INFORMATION ON NUCLEAR SAFETY MANAGEMENT AND STA

DEPARTMENT: LICENSING

AS OF: 8/31/81

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
G. C. Sorensen	BS Mechanical Engineering	Manager, Licensing	8 yr./7 yr./15 yr.	Licensing	Mechanical Engineering	Amphenol Sams Division	Project Engineer
R. M. Nelson	BS Mechanical Engineering Nuclear Option	Manager, WNP-2 Project Licensing	2 mo./21 yr./21 yr.	Nuclear Eng. Mech. Eng. Design	NSSS Design NSSS Licensing Chemical Eng.	General Electric Co.	Senior Licensing Engineer
A. G. Hosler	BS Electrical Eng. MS Nucl. Eng.	Manager, WNP-1/4 Priject Licensing	5 mo./18 yr./18 yr.	Licensing Nucl. Eng.	Analysis	Westinghouse	Senior Development Engineer
✓ K. W. Cook	BS Mechanical Eng. Nuclear Option MS Mechanical Eng. ABC Courses (GE)	Manager, WNP-3/5 Project Licensing	16 mo./20 yr./21 yr.	Nucl. Eng. Licensing	Mech. Eng. Safety Eval.	General Electric Co.	Senior Licensing Engineer
K. A. Hadley	BS Mechanical Eng. BS Industrial Eng. Naval Nuclear Power School	Engineer I	1 mo./15 yr./15 yr.	Mechanical Eng. Industrial Eng.	Operations/Maint./ Systems Generic Issues	Westinghouse	Cognizant Design Engr.
✓ R. G. Joshi	BS Electrical Eng. MS Nuclear Eng.	Senior Engineer	2 mo./14 yr./14 yr.	Licensing Eng. Nuclear Eng.	I&C Eng. Electrical Eng.	Stone & Webster Energy Corp.	Licensing Engineer
P. L. Powell	BS Electrical Eng. Naval Nuclear Power School	Engineer I	4 yr./7 yr./11 yr.	Nuclear Eng. Electrical Eng.	Safety Analysis Administrative Eng.	Westinghouse	Admin. Engineer
A. J. Moore	BS Physics Grad. Studies in Meterology	Project Engineer I	2 yr./12 yr./14 yr.	Operations at FFTF (Operator Training) NDE/NDT Training Lead Auditor	Quality Assurance	Westinghouse	Reactor Operator

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✓ Registered P. E., other state(s).

FORMATION ON NUCLEAR SAFETY MANAGEMENT AND STAF

DEPARTMENT: LICENSING

AS OF: 8/31/81

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
C. D. Taylor	AA	Licensing Assistant, WNP-2	8 yr./ /8 yr.	FSAR Amendments	NRC Questions	------	------

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✓ Registered P. E., other state(s).

DEPARTMENT: OPERATIONAL NUCLEAR SAFETY

AS OF: 8/31/81

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
Q. L. Baird	AB Mathematics MS Physics PhD Physics	Manager, Operational Nuclear Safety	6 mo./23 yr./24 yr.	Nuclear Eng. Reactor Physics	Nucl. Safety Tech. Reactor Operations	Westinghouse	Manager, Operational Nuclear Sfty
C. H. McGilton	MS Metallurgy ORNL School of Reactor Tech. Licensed SRO on Cooper	Manager, SEG WNP-2	8 yr./10 yr./18 yr.	Reactor Operations Operations Supv. at WNP-2 (7 yr.)	Nuclear Eng.	Northern States Power	Operations Engineer
S. C. Denison	BS Mechanical Eng.	Manager, SEG WNP-1/4	8 yr./9 yr./17 yr.	Nuclear Eng. Mechanical Eng.	Reactor Operations	Westinghouse	Senior Proj. Engineer
D. W. Coleman	BS Mechanical Eng. MS Mechanical Eng. Grad. work in Nuclear Eng.	Manager, SEG WNP-3/5	6 mo./12 yr./13 yr.	Mechanical Eng.	Nuclear Eng.	U. S. Dept. of Energy	Nuclear Eng.
F. D. Frisch	BS Mechanical Eng. Licensed SRO at Millstone 1 & Pilgrim	Principal Engineer	6 yr./15 yr./21 yr.	Mechanical Eng. Reactor Operations	Start-Up Testing	General Electric Co.	Start-Up Operations Super-intendent
R. G. DaValle	BS Engineering	Senior Engineer	8 yr./7 yr./15 yr.	Reactor Operations Control Room Human Factors Review	General Eng.	Bechtel Power	Sr. Field Scheduling Engineer
A. K. Yee	BS Physics MS Physics PhD Physics Grad. work in Nuclear Eng.	Principal Engineer	4 mo./31 yr./31 yr.	Nuclear Eng. Reactor Analysis Safety Analysis	Nuclear Safety Tech.	Westinghouse	Principal Eng

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INFORMATION ON NUCLEAR SAFETY MANAGEMENT AND STAFF

DEPARTMENT: OPERATIONAL NUCLEAR SAFETY

AS OF: 8/31/81

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
✓ M. D. Zentner	BS Physics MS Nuclear Eng.	Senior Engineer	1 mo./8 yr./8 yr.	Nuclear Eng. Physics	Reactor Analysis Reactor Operations	Brookhaven National Lab.	Research Engineer
+ R. M. Norris	BS Mechanical Eng. MS Mechanical Eng.	Principal Engineer	5 yr./10 yr./15 yr.	Mechanical Eng.	Nuclear Eng. NSSS Design	General Electric Co.	Program Manager

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✓ Registered P. E., other state(s).

INFORMATION ON NUCLEAR SAFETY MANAGEMENT AND STAFF

DEPARTMENT: DESIGN AND NUCLEAR SAFETY ASSESSMENT

AS OF: 8/31/81

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
S. L. Additon	BS Mathematics MS Nuclear Eng. BA	Manager, Design & Nuclear Safety Assessment	7 mo./13 yr./14 yr.	Thermal Hydraulic & Plant Dynamic Analysis	Nuclear Safety	Westinghouse	Manager, Systems Design
✓ J. W. Sale	BS Mechanical Eng MS Mechanical Eng Reactor Operation	Manager, Desing Assessment	3 yr./18 yr./21 yr.	Mechanical Eng.	BOP System Design Fire Protection	Pacific Gas & Electric Co	Mechanical Engineer
R. O. Vosburgh	BA Physics/Math MS Physics	Manager, Nuclear Safety Analysis	1 yr./16 yr./17 yr.	Nuclear Eng. Transient Anal. (T-H Neutronic)	Risk Assessment	Babcock & Wilcox	Senior Engineer
S. M. Baker	BS Mathematics MS Nuclear Eng. PhD Mathematics	Manager, Nuclear Safety Standards & Technology	7 mo./16 yr./17 yr.	Mathematics Nuclear Eng.	Reactor Operations	U. S. Dept of Energy-FFTF	Safety & Operations Staff
C. J. Foley	BS General Eng. MS Nuclear Eng.	Principal Engineer	7 mo./15 yr./16 yr.	Mechanical Eng.	Nuclear Plant Start-Up Supv. (1975-1980)	Westinghouse	Principal Engineer
✓ R. A. Hazen	None	Senior Engineer	6 mo./19 yr./20 yr.	BWR Systems	BOP	General Electric	Senior Project Eng.
E. D. Shoua	BS Electrical Eng. MS Structural Eng. PhD Material & Applied Mech.	Senior Engineer	4 yr./11 yr./15 yr.	Nuclear Fuel Safety Analysis	Material & Stress Analysis Structural Design	Nuclear Services Corp.	Nuclear Fuel Consultant
+ J. M. Henderson	BS Nuclear Eng.	Engineer I	1 mo./9 yr./9 yr.	Nuclear Eng. Instrumentation	Computer Systems Nuclear Fuel	Westinghouse	Engineer I

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INFORMATION ON NUCLEAR SAFETY MANAGEMENT AND STAFF

DEPARTMENT: DESIGN AND NUCLEAR SAFETY ASSESSMENT

AS OF: 8/31/81

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
P. R. Shire	BS Math/Chem/ Physics MS Nuclear Eng.	Principal Engineer	5 mo./27 yr./27 yr.	Nuclear Eng. LWR/LMFRB Safety Analysis Neutronics/Thermal Hydraulics Ana.	Plant Heat Balance Generation Planning Computer Code Spec.	Westinghouse	Reactor Safety Eng.
J. S. Dukelow	AB Mathematics MA Mathematics MS Nuclear Eng.	Senior Engineer	4 mo./11 yr./11 yr.	Nuclear Eng. Thermal Hydraulics	Reliability Mathematics	HEDL	Reliability Engineer
D. L. Prezbindowski	BS Engr. Science MS Nuclear Eng. PhD Nuclear Eng.	Senior Engineer	5 mo./13 yr./13 yr.	Mechanical Eng. Nuclear Eng.	Core Physics & Shielding Calculations	Battelle	Sr. Patent Engineer
✓ B. L. Twitty	BS MS Radiochemistry MS Nuclear Eng. Certified in Health Physics	Principal Engineer	3 mo./25 yr./25 yr.	Reactor Control Inst. Startup - Accept.	Safety Anal., Health Physics, Plant Design, & Radiochemistry	Westinghouse	Senior Engineer

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RMATION ON QUALITY ASSURANCE MANAGEMENT AND STA

DEPARTMENT: Quality Assurance

AS OF: September 8, 1981

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
Casavant, R. D.	(A.S.) Business Administration	QA Engineer I	3 mos/2.5 yrs/2.75 yrs.	Mechanical Systems	Business Admin.	Crosby Valve & Guage Co.	QA Supvr.
✓ Chandon, H. C.	(B.S.) Electrical Engineering	Senior QA Engineer	3.5 yrs/19.5 yrs/ 23 yrs	QA Engineer	Electrical Engineer	Bechtel Power Corp.	Sr. QA Engr.
✓ Dunn, T. E.	(B.S.) Mechanical Engineering	Principal QA Engineer	1 mo/22 yrs/22 yrs	QA Program Development & Maintenance	Fuel Fabrication QA Programs	Battelle	Sr. Scientist
Feldman, D. S.	(B.S.) Industrial Engineering	QA Engineer I	1 yr/5 yrs/6 yrs	QA Operations	Power Plant Operations	U. S. Navy	QA Engineer
✓ Garvin, L. J.	(B.S.) Chemical Engineering (B.S.) Nuclear Engineering) (B.S.) Metal. Engineering	Manager, Quality Performance & Measurements	2 yrs/17 yrs/19yrs	QA & Metallurgical Engineering	Nuclear Safety	USNRC	Reactor Inspector
Gross, V. R.		QA Engineer I	4 yrs/14 yrs/18 yrs	Mechanical/Elect.	I/C, Civil	UE&C	QA Supervisor
Gupta, S. L.	(B.S.) Physics	Principal QA Engineer	2.5 yrs/17.5 yrs/ 20 yrs	Quality Assurance	a. Fueling & Re-fueling Operations b. Nuclear Plant Operations	Carolina Power & Light	Senior QA Engineer
Henthorn, C. O.		Senior QA Engineer	1.5 yrs/31.5 yrs/ 33 yrs	Quality Assurance	Business Management	Self	QA Consultant
Houchins, T. J.		Manager, Audits and Surveillance	7 yrs/19 yrs/26 yrs	Quality Assurance	Auditing	UE&C	Site Surveillance Supervisor

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✓ Registered P. E., other state(s).

DEPARTMENT: Quality Assurance AS OF: September 8, 1981

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
Kooy, W.	(B.S.) Electrical Engineering	Principal QA Engineer	4.5 yrs/16.5 yrs/ 21 yrs	Test & Startup Electrical, Instrument, & Control	Chemical Analysis	MATSCO	Director of Containment Testing
✓ Krolicki, R. P.	AA-Radiogr. BA-Commerce MA-Economics	Principal QA Engineer	4 yrs/23 yrs/27 yrs	Quality Assurance, NDE	Welding	Florida Power & Light Co.	Senior QA Engineer
Lauck, J. A.	(B.S.) Accounting	QA Engineer I	1 mo/12 yrs/12.1 yrs	Mechanical-equip./ piping erection & testing	Welding/NDE	WBG #2	Procurement Engineer
Simons, R. M.		Manager, Quality Systems	1.5 yrs/22 yrs/ 23.5 yrs	QA Systems	NDE	Richland Engineering	QA Manager
Walker, J. M.		Senior QA Engineer	6 yrs/5.5 yrs/11.5 yrs	QA Surveillance/ Audits	QA Systems/Programs	Northeast Utilities Service Co.	Engineering Technician
Webster, R. D.	(BA) Education	QA Engineer I	6 mos/39.5 yrs/ 40 yrs	Systems Mechanical	Electrical	Westinghouse	QA Engineer
Winters, H. L.		QA Receiving Inspector	6 yrs/21 yrs/27 yrs	Receiving Inspection	Mechanical, Dimensional Measurements	HUICO	QA Inspector
✓ Wooley, G. H.		Manager, Vendor QA	4.5 yrs/5.5 yrs/ 10 yrs	QA General Administration, Supervisory	Mechanical, Welding, NDE	Bechtel	Vendor Surveillance Engineer

+ Registered P. E., Washington.
✓ Registered P. E., other state(s).

RMATION ON QUALITY ASSURANCE MANAGEMENT AND STA

DEPARTMENT: Quality Assurance

AS OF: September 8, 1981

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
✓ Zimmerschied, J. R.	(B.S.) Physics Engineering	Senior QA Engineer	3 yrs/9 yrs/12 yrs	QA Surveillance & Audits in piping & mechanical area	Programming and use of micro-processors	Bechtel Power Corp.	Project QA Engineer

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✓ Registered P. E., other state(s).

INFORMATION ON WNP-2 MANAGEMENT AND STAFF

DEPARTMENT: WNP-2 PROGRAM DIRECTOR

AS OF: 9-1-81

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
R.G. Matlock	BSME PH.D. Physics	Program Director, WNP-2	1 yr/20 yr/21 yr	Nuclear Reactor Design, Operations, Project Management	Nuclear Physics, Research & Development.	U.S. DOE, ANL	Deputy Director, for Nuclear Programs, Physicist, Program Director, Project Mgr

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√ Registered P. E., other state(s).

INFORMATION ON WNP-2 MANAGEMENT AND STAFF

DEPARTMENT: WNP-2 PROGRAM DIRECTOR

AS OF: 9/1/81

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
C.S. Carlisle	BS Naval Science MS Business Admin	Deputy Program Director, WNP-2	4 mos/35 yrs/35 yrs	Nuclear plant operations and maintenance	Construction Project Management	U.S. Navy	25 yrs experience in nuclear submarine construction, operations, & maintenance.

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√ Registered P. E., other state(s).

DEPARTMENT: PROGRAM CONTROL

OF: September 4, 1981

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
W.D. Holloman	B.S. Eng. (Gen) (U.S. Naval Academy) B.S. Eng. (M.E.) Nuclear Power	Executive Assistant Mgr. Program Control (Acting)	6 mos/26yrs/26yrs	Resource Mgmt.	Tech Management (Nuclear Pwr R&D)	AEC/DOE 6 yrs. U.S. Navy 20 years	(Budget Officer) (Line Officer)

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✓ Registered P. E., other state(s).

INFORMATION ON WNP-2 MANAGEMENT AND STAFF

DEPARTMENT: Programs's Directors

AS OF: September 4, 1981

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
J.R. Honekamp	B.S. & M.S. Chem Eng. PH.D Chem Eng. with minor in Nuclear Eng.	Technical Specialist	10 mo./20/21	Design and Testing of Nuclear Power Plants	Fuels and Materials Performance, and Operations	Argonne National Lab and Knolls Atomic Power Lab.	11 years experience in design and testing associated with Breeder Reactor Program 9 years experience in design, testing and operation of Naval Power Plants

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✓Registered P. E., other state(s).

INFORMATION ON WNP-2 MANAGEMENT AND STAFF

DEPARTMENT: Project Quality Assurance

AS OF: 9-4-81

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
Roger T. Johnson	None	WNP-2 Project QA Manager	WPPSS 8 years QA Mngmt. functions. 9 years experience in Nuclear Const. and Project Mngmt. 17 years total experience.	Nuclear Const. and Project Mgmt.	Mechanical systems design, installation and inspection.	Westinghouse Hanford	Project Engineer
Richard E. Spence	B.S. Nuclear Engineering	QA Compliance Supervisor	1½ yrs/14yrs/15yrs	Nuclear power plant const. and Quality Assurance.	Nuclear power plant test, startup and operations. Also, radiation protection	Mid-Columbia Engineering Co.	Project QA Engineer
Carl O. Wright	BSME	QE Supervisor	3yrs/7½yrs/10½yrs	Nuclear power plant const. Quality Assurance Audits/Surveillances	Mechanical Engineering with power utilities.	Bechtel 4½yrs.	SR QAE

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INFORMATION ON WNP-2 MANAGEMENT AND STAFF

DEPARTMENT: OPERATIONAL QUALITY ASSURANCE

AS OF: 9/8/81

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
Joseph M. Graziani	B.S. Materials Engineering, 6/71	Operational QA Supervisor WNP-2	2 yrs 10 mo. w/Supply System 6 yrs 4 mo w/Bechtel 9 yrs. 2 mo - Total	Nuclear Construction Field Engineer, (startup System completion & Electrical check-out)	Operational Quality Assurance Engineer -Nuclear Management Plan for PRE-OP and operation phase	Bechtel Power Corp.	Electrical Field Engineer Startup System Turnover Administrator Construction Completion Coordinator Lead Field Cost Engineer Field Cost Engineer

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Registered P. E., other state(s).

INFORMATION ON WNP-2 MANAGEMENT AND STAFF

DEPARTMENT: SUPPORT SERVICES

AS OF: September 1, 1981

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
J. A. Harrington	BA Business Administration	Manager, Support Services Division	7 yrs/18 yrs/25 yrs	Business Management	Quality Assurance Management	Bunker-RAMO Corporation	Reliability & Quality Control Manager

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√ Registered P. E., other state(s).

INFORMATION ON WNP-2 MANAGEMENT AND STAFF

DEPARTMENT: Administration

AS OF: September 3, 1981

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
J. F. Peters	BA Psychology	Administration Manager WNP-2	5 yrs./6 yrs./11 yrs.	Commercial Nuclear Plant Administration & Support Services	Operations and Startup Management	Metropolitan Edison Co.	Station Administrator

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INFORMATION ON WNP-2 MANAGEMENT AND STAFF

DEPARTMENT: WNP-2 RECORDS MANAGEMENT

AS OF: September 1, 1981

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
T.V. Anderson	BA History	Manager, Records Management	4yrs/3 yrs/7 yrs	Records Management	Archives Administration	National Archives & Records	Records Management Intern

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✓ Registered P. E., other state(s).

INFORMATION ON WNP-2 MANAGEMENT AND STAFF

DEPARTMENT: Project Materials Management

AS OF: September 1, 1981

NAME	DEGREE	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPLE AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
D. R. Herling	AA Business Admin.	Manager, Project Materials Management, WNP-2	4 yrs/15 yrs/19 yrs	Procurement & Materials Mgmt. Process Plants & Nuclear Construction	Manufacturing	Davy McKee	Project Material Coordinator

INFORMATION ON WNP-2 MANAGEMENT AND STAFF

DEPARTMENT: Business

AS OF: September 1, 1981

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
R. E. Baker	BS Management & Military Science MS Mech. Engineering	Business Manager	4½ yrs/23 yrs/23 yrs	Business	Construction Mgmt.	CSTDC/Great American Ins. Company	Consultant

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INFORMATION ON WNP-2 MANAGEMENT AND STAFF

DEPARTMENT: OPERATIONS

AS OF: 9/4/81

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
J. D. MARTIN	BS Industrial Management Brigham Young 1960	WNP-2 Plant Manager	3 / 18 / 21	Reactor Operations	BWR Startup	US NRC	Reactor Inspector
						General Elec. Browns Ferry	Ops. Mgr. Ops. Supt. Training Inst. Startup Test. Engr.
						GE-Hanford	Reactor Ops Supervisor
						DUN	Reactor Oper Supervisor

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INFORMATION ON WNP-2 MANAGEMENT AND STAFF

DEPARTMENT: TRAINING

AS OF: 9/4/81

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
R. D. DAVIDSON	BS in Science Education	WNP-2 Training Supervisor	6 / 9 / 15	Development and Managing Training Programs	Reactor Operation	General Elect.	Simulator Training Engineer
						US Navy	Reactor Oper/ Training

+ Registered P. E., Washington.
√ Registered P. E., other state(s).

INFORMATION ON WNP-2 MANAGEMENT AND STAFF

DEPARTMENT: Plant Technical

AS OF: September 4, 1981

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
K. D. Cowan Registered Prof. Engineer, Mech.-California	1) BS-Mech. Eng. 2) Masters-Bus. Administration	WNP-2 Plant Technical Superintendent	WPPSS-WNP-2 Project Engineering Manager 1½ years WPPSS-WNP-2 Plant Technical Supt. 1 year GE-Nuclear Energy Division 1) Project Eng. 2) Systems Eng. 3) Design Eng. 17 years total	Technical or Project Engineering Management	Mechanical Systems Engineering	General Electric - Nuclear Energy Division	Various- from Design Engineer to Project Engineer to Systems Engineer to Unit, Subsection and Section Manager

+ Registered P. E., Washington.
(√) Registered P. E., other state(s).

INFORMATION ON WNP-2 MANAGEMENT AND STAFF

DEPARTMENT: Operations/Maintenance

AS OF: September 3, 1981

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
R. L. Corcoran	B.S.E.E.	Plant Operations Superintendent	8/7/15	Nuclear Plant Operations	Plant Technical/ Engineering	Dairyland Power Coop. LaCrosse BWR Gen. Station	Plant Tech. Supervisor/ Project Engr
J. W. Hedges	B.S.M.E. M.S. Geology	Plant Maintenance Supervisor	8/23/31	Nuclear Plant Maintenance	Plant Technical/ Engineering	UNI, N-Reactor	Sr. Engr. N-Reactor Maintenance

+Registered P. E., Washington.
✓Registered P. E., other state(s).

INFORMATION ON WNP-2 MANAGEMENT AND STAFF

DEPARTMENT: Project Management

AS OF: 9/4/81

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
W. C. Bibb	3 yrs towards EE degree Registered P.E., California	Project Manager, WNP-2	5 yrs/22 yrs/ 27 yrs	Startup/Ops/Power Production Mgr	Project Management	General Elect.	Area Mgr Nuclear Services Midwest Reg. Site Mgr Cooper Nuc. Station

+ Registered P. E., Washington.
√ Registered P. E., other state(s).

INFORMATION ON WNP-2 MANAGEMENT AND STAFF

DEPARTMENT: Test & Startup

AS OF: ____

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
G. K. Afflerbach	BS Electrical Engineering	Deputy Project Manager, Startup, WNP-2	3 yrs/14.5 yrs/17.5 yrs	Nuclear plant startup, operation and maintenance	Nuclear plant retrofit & outage management	General Electric	Operations Manager

+ Registered P. E., Washington.
✓ Registered P. E., other state(s).

INFORMATION ON WNP-2 MANAGEMENT AND STAFF

DEPARTMENT: Construction Management

AS OF: September 4, 1981

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
H.A. Crisp	B.S. Mining Eng. MS Petroleum Eng. MS Civil Eng.	Deputy Project Manager Construction	1 yr/25yrs/26yrs	Management of Plant Construction and Maintenance	Engineering	U.S. Navy	25 years experience in facilities maintenance and const. last 7 yrs. associated with Nuclear or Nuclear Support Facilities

+ Registered P. E., Washington.
✓ Registered P. E., other state(s).

INFORMATION ON WNP-2 MANAGEMENT AND STAFF

DEPARTMENT: WNP-2 Project Engineering 62900

AS OF: 9/4/81

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
BA Holmberg	BS Marine Engrg.	Deputy Project Manager-Engineering	2.75/20/22.75	Management/Nuclear/Operations	Mechanical	U.S. Navy	Commander/Navy

+ Registered P. E., Washington.
✓ Registered P. E., other state(s).

INFORMATION ON WNP-2 MANAGEMENT AND STAFF

DEPARTMENT: Project Finance

AS OF: September 8, 1981

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
L.R. Travis	B.S. Business Administration	Project Controller, WNP-2	3 yrs./31 yrs./34 yrs.	Construction Financial Management	Manufacturing Financial Management	Kaiser Engineers	Administrative Manager
C.W. Robinson	B.S. Commerce & Finance	Project Accounting Manager, WNP-2	BRI-12 yrs./11 yrs./ 23 yrs.	Construction Accounting Mgt.	Construction Administration	The J. G. White Engineering Corp.	Field Office Manager
J.R. Dufort	B.S. Business Administration	Project Budget Manager	2 yrs./37 yrs./39 yrs.	Budget & Cost Accounting	Cost Engineering	General Electric Company	Senior Cost Engineer

+ Registered P. E., Washington.
✓ Registered P. E., other state(s).

INFORMATION ON POWER GENERATION MANAGEMENT AND [illegible]

DEPARTMENT: POWER GENERATION DIRECTORATE

AS OF: August 31, 1981

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
DEEGAN, G. E.	BS-Math & Physics	Manager, Generation Mgmt. Services	1.5/29.5/31	Mgmt. of Reactor Operations/Maint.	Nuclear & Liquid Metal R&D Mgmt.	Argonne Nat'l. Lab	EBR-II Operations Mgr.
HANNAH, K. J.	BS-Welding & Metallurgical Engineering	Supv., Nondestructive Examination & Inspection	4/15/19	Nondestructive Examination; Preservice & Inservice NDE Evaluation; Registered Quality Engineer	Failure Analysis in BWR Systems	GE; Automation Industries; Lockheed Missiles & Space; Rocketdyne	Sr. Engr.; Mat'ls. & Chemistry; Research Engr.; Mfg. Engr.; Field Engr.
HOLDER, J. R.	BS-Math & Engineering	Program Manager, Power Generation	2/9/11	Nuclear Power Plant Operations/Startup & Testing/Management; Operational Design Review; Startup & Operational Programs Development & Management		Carolina Power & Light	Plant Supt.; Plant Startup Engr.; Sr. Engr.; Refueling Engr.; Nuclear Engr.; Sr. Reactor Engr.
HOLZENDORF, W. F.	HS; 1 Yr. College; Navy Nuclear Power School; Navy Engineering Officer School	Acting Supv., Maintenance Services (Sr. Maintenance Specialist)	1/23/24	Mechanical/Systems	Operations Engineer	Westinghouse Hanford; U.S. Navy	Operation Support Engr.; Seaman
ITTNER, J. P.	BS-Marine Engineering	Principal Engineer	6.5/9.5/16	Nuclear Operations & Startup; Training Simulator Project Management.	Reactor Operations; Nuclear Technology/Systems Trng.; Planning & Scheduling	Babcock & Wilcox; First Atomic Ship Transport, Inc	Site Operations Engr./Technical Support Supv.; Reactor Operator
KUBENKA, J. H.	BS-Occupational Education	Supv., Support Training	1/15/16	Reactor Operations/Maintenance/Training; Project Engr.	Computer Programming & Statistical Analysis	United Nuclear Corp.; US Navy	Trng. Rep.; Nuclear Power School Advisor

Registered P. E., Washington.
Registered P. E., other state(s).

DEPARTMENT: POWER GENERATION DIRECTORATE

AS OF: August 31, 1981

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
McLEOD, B. K.	Navy Nuclear Reactor Program	Manager, Power Generation Services	8/12/20	Nuclear Power Plant Operations; Quality Assurance	Nuclear/Industrial Safety; Training	United Nuclear Corp.; NRC; General Electric Co.; U.S. Navy	Safety/Licensing Mgr.; Principal Reactor Insp.; Shift Mgr.; Reactor Operator
NOYES, C. R. †√√	BSME; MSNE; Registered Professional Engr. (3 states)	Manager, Power Generation Maintenance	2 Mos./24/24	Mechanical Engineer; Nuclear Engineer; Nuclear & Mechanical Maintenance; Reactor Operations; Navy Nuclear Programs	Master Precision Machinist; Special Tool Design	U.S. Navy; U of Nebraska; Nebraska Public Power Dist.; U of Missouri	Nuclear Lab Tech./Machinist; Nuclear Engr. Asst.; Engrg. Supv.
PARTRICK, R. E.	BS-Naval Science; Navy Nuclear Power School	Supv., Generation Coordination	6.5/6.5/13	Reactor Operations/ Maintenance	Maintenance Engrg. Mgmt.	Westinghouse Hanford	Maint. Dev. Engr. Sr.
PERRY, J. F.	65 Units of College-Math & Engineering	Supv., Simulator Training Systems	5/10/15	Electronics Technician; Simulator Procurement & Operations	Previously licensed as SRO on Indian Point	Con Edison of New York; U.S. Navy	Sr. Simulator Instructor; Reactor Operator
ROSENECK, J. B.	USN-ETA/ETV/NPTU; GE-STE/BWRTC	Supv., Operational Training	2.5/18.5/21	Nuclear Analyst; Reactor Plant Engineering; Reactor Core Design & Analysis; Plant Operation; Shift Test Engr.; Shift Supv.,; USN/RO/RT; Startup Engineer; Supv., Turnover	Computer Programming Electronic Maintenance	GE-NEBG & KAPL U.S. Navy	Shift Supv.; Shift Test Engr.; Reactor Control LPO

Registered P. E., Washington
Registered P. E., other state(s).

INFORMATION ON POWER GENERATION MANAGEMENT AND STAFF

DEPARTMENT: POWER GENERATION DIRECTORATE AS OF: August 3, 1981

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
TERRASS, M. S. ✓	BS-Naval Science; MBA; Navy Advanced Nuclear Power School	Principal Engineer	3.5/17/20.5	Nuclear Engineering; PE/Nuclear; Education & Training; Nuclear Plant Operations (BWR & PWR)	Nuclear Plant Testing & Maintenance (BWR & PWR)	U.S. Navy; IBM; Nuclear Services Corp.; California Energy Commission	Naval Officer; Data Processing Marketing Rep; Specialist Engr; Energy Facility Site Planner
CHUBB, K. L.	Technical Institute Graduate; GE-Computer Maintenance	Supv., Instrument Maintenance & Calibration	3/23/26	Nuclear Plant Support	Computer Maintenance; Standards Laboratory Supervision; Instrument Technical Instruction	Westinghouse Hanford; Battelle NW; General Electric	Calibration & Computer Maintenance Mgr.; Computer Maintenance Spec.; Computer Field Engr.; Instrument Technician

Registered P. E., Washington. †
Registered P. E., other state(s). ✓

INFORMATION ON TECHNOLOGY MANAGEMENT AND STA[illegible]

DEPARTMENT: TECHNOLOGY

AS OF: 7/31/81

	NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
+	P. K. Shen	BS Chemical Engr. MS Chemical Engr. PhD Nuc. Engr.	Director, Technology	1 yr./14 yr./15 yr.	Reactor Analysis, Nuclear Systems Design, Nuclear Reactor Safety, Nuclear Fuel Mgmt.	Reactor Operation, Licensing, Nuclear Waste, Radiation Effect Materials	Univ. of Washington	Dean and resident Director, Professor Nuclear Engineering
	D. L. Renberger	BS Nuclear Engr.	Deputy Director, Technology	10 yr./13 yr./23 yr.	Nuclear Plant Systems Performance and Testing	Nuclear Safety and Licensing	Douglas United Nuc.	Supervisor Regulatory Unit
✓	L. L. Grumme	BS Nuclear Engr.	Assistant Director, Technology	10 yr./11 yr./21 yr.	Nuclear Fuel procurement Nuclear Fuel Mgmt. Management	Reactor Operations Environmental Licensing Nuclear Safety	Douglas United Nuc.	Sr. Engr.
†✓	J. N. Salapatas	BS Industrial Engr. MS Management	Manager, Technical Admin.	1 yr./25 yr./26 yr.	Project Management Proj. Controls/ Nuclear Engr. & Construction	Management Systems, Organization Planning and Development and Training	Florida Power & Light	Manager, Project Control Services
	D. W. Fraley	BS Engineering Sciences PhD Systems Engineering	Administrator, Technical and Information Systems	4 mo./12 yr./12 yr.	Systems Analysis, Decision Analysis, Operations Research Applied to Nuc. Industry	Computer Science, Information Systems, Hardware & Analysis	Battelle	Sr. Research Scientist, Nuclear Energy Systems

✓ Registered P.E., Washington.
† Registered P.E., other state(s).

INFORMATION ON ENGINEERS IN ENGINEERING DIVISION

DEPARTMENT: Engineering Division Staff (42100)

AS OF: 7-31-81

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS	OTHER	TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
CL Fies	BSEngPhys, MSNE, MBA	Program Mgmt Spec	7	10	17	Program Mgmt	-----	Westinghse	Sr Engr
✓ LT Harrold	BSME, BAEcon	Engr'g Division Manager	8	8	16	Mechanical	ISI	DUN	Sr Engr

✓ Registered P.E., Washington.
† Registered P.E., other state(s).

INFORMATION ON ENGINEERS IN ENGINEERING DIVISION

DEPARTMENT: Mechanical Engineering (42500) AS OF: 7-31-81

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS	EXPERIENCE OTHER	EXPERIENCE TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
✓ WD Bainard	BSChE	Consulting Engr	9.25	20	29.25	Corrosion	Water Quality/N-C	DUN	Sr Engr
D Burns	BSMetl, PhD	Supv, Materials & Components	3.75	5	8.75	Metallurgy	Welding	IL Rsch Inst	Rsch Mtlgst
†DM Bosi	BSME, MSME	Sr Engr	0	9	9	Engr'g Mechanics	Heat Transfer	Westinghse	Sr Engr
MK Chakravorty	PhD/Str.Dyn.	Principal Engr	2.75	7	9.75	Earthquake Engr'g	Structural Dynamics	Stone/Webstr	Prin Engr
JR Cole	BSME	Engr I	.5	9	9.5	Machine Design	-----	WSH	Sr Engr
DW Cutting	BSMetE	Principal Engr	1.75	13	14.75	Welding	Metallurgy	WGC	Chief Wldg Eng
TM Erwin	BSMetE	Associate Engr	.5	0	.5	Weld'g/Metallurgy	Welding processes	MI NavShpyd	Engr in Trng
✓†HE Good	BSME, MSCE	Supv, Engr'g Mechanics	3	17	20	Structural Dynamics	Seismic/Soil Mech.	Stone/Webstr	Lead Engr
TF Hoyle	BSME	Senior Engr	3	7	10	ISI/PSI/NDE	Welding	IL Pwr	ISI Engr
JD Husted	AAS	Engr'g Assistant	1.75	.25	2	Engr'g Documentn	-----	UNI	Lab Tech
PJ Inserra	BSEngTech	Associate Engr	1	.25	1.25	Welding	Metallurgy	DiabloCanyon	QA Inspctr
RJ Lauzon	BSME	Senior Engr	2	20	22	Welding	Piping	Consultant	Self-Employd
✓ RL Loundagin	BSME	Senior Engr	3.5	24	27.5	Piping (ASME)	Stress Analysis	UNI	Sr Engr
RA Moen	BSMetE	Senior Engr	1.5	18	19.5	Metallurgy	ASME	Westinghse	Sr Engr
†EL Morales	BSME	Senior Engr	3.75	13.5	17.25	Stress Analysis	Piping	Stearns-Rgrs	Engr III
✓ JC Mowery	BSME	Principal Engr	8.75	12	20.75	Piping	Systems Engr'g	HUICO	Proj Engr
DA Murdock	BSME	Engr I	.5	10.5	11	NDT (Ultrasonics)	-----	NucEngySvcs	NDE Engr

✓ Registered P.E., Washington.
† Registered P.E., other state(s).

INFORMATION ON ENGINEERS IN ENGINEERING DIVISION

DEPARTMENT: Mechanical Engr'g - CONTINUED (42500)

AS OF: 7-31-81

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS	OTHER	TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
✓ JE Parker	BSME	Senior Engr	5.5	6	11.5	Rotating Equipt	Systems Engr'g	GE	Serv Engr
✓†DW Porter	BSNE	Manager, Mechanical Engr'g	4.75	3.5	8.25	ISI/PSI/NDE	Systems Engr'g	Bechtel	Sr Nuc Engr
LR Prill	BSME	Engr I	.25	9	9.25	Ops/Maint/Design	Personnel Trng	US Navy	Sub Officer
DP Ramey	BSChE	Engr I	.75	5	5.75	ISI/PSI	-----	Rockwell	Plt Engr
†MP Reis	BAMath, MSNE	Engr I	1.25	4	5.25	Systems Engr'g	ISI/PSI/NDE	PG&E	T&SU Engr
CD Scott	BSME	Engr II	1.25	3	4.25	Piping Systems	Heat Transfer	WSH	Engr
K Singh	BSME	Senior Engr	4.25	8.5	12.75	ISI/PSI/NDE	Piping	Bechtel	Pipg Engr

✓ Registered P.E., Washington.
† Registered P.E., other state(s).

INFORMATION ON ENGINEERS IN ENGINEERING DIVISION

DEPARTMENT: Civil Engineering & Services (42600)

AS OF: 7-31-81

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS	EXPERIENCE OTHER	EXPERIENCE TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
PJ Adair	None	Sr Engr'g Records Analyst	6.25	0	6.25	Documentation	-----	-----	-----
AR Bright	BCS	Sr Admin Engr	4.75	15	19.75	Engr'g Admin	Accounting	Boeing	QA Supv
KM Brun	BASecEduc	Engr'g Records Analyst II	1.5	9.25	10.75	Equipt Qualif	Documentation	CPCO	Plt Clerk
BM Burnside	None	Engr'g Records Analyst I	1.25	4	5.25	Documentation	Comp Applic	B&R	Technician
PE Campbell	ASNucTech	Engr'g Records Analyst II	0	3	3	-----	-----	Westinghse	Technician
KN Chick	BSAnthrp	Engr'g Records Analyst I	3.5	3	6.5	Equipt Qualif	Comp Applic	-----	-----
MA Coffman	AA	Engr'g Records Analyst II	0	4	4	Documentation	-----	Westinghse	Technician
CR DePoe	None	Senior Engr	3.25	21.5	24.75	Mechanical	Electrical	SD St Univ	Manager
VK Duggal	BAArcht	Senior Designer	2	11	13	Design/Drafting	-----	B&R	Designer
✓†WC Flynn	BSCE	Senior Engr	2.75	21	23.75	Fac Des/Install	Struct Design	Pt/Olympia	Chief Engr
FV Gomulka	BA Hum.Resc	Engr'g Records Analyst I	1.25	1.5	2.75	Documentation	Comp Applic	Westinghse	Data Clerk
WS Hardy	None	Engr'g Records Analyst II	2.25	3	5.25	Documentation	-----	Bechtel	Elect Clerk
NH Kempt	BSEE	Senior Engr	3.25	18	21.25	Electrical	-----	Vitro	Sr Engr
WH Knolle	None	Engr I	3.5	37	40.5	Elect Eng/Design	Cost Estimating	Vitro	Sr Estmtr
LE Maravilla	None	Admin Specialist	7.75	4	11.75	Administration	Secretarial	Juv Court	Sec'y
KF McAndrew	AAS Tech Illust	Engr'g Asst (Drafting)	3.75	7	10.75	Design/Drafting	-----	Bovee/Crail	Drafter

✓ Registered P.E., Washington.
† Registered P.E., other state(s).

INFORMATION ON ENGINEERS IN ENGINEERING DIVISION

DEPARTMENT: Civil Engr'g & Svcs - CONTINUED
(42600)

AS OF: 7-31-81

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS	OTHER	TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
BM Nichols	None	Sr Engr'g Records Analyst	5.5	17	22.5	Documentation	Engr'g Admin	City/Rchld	Sec'y
EV Norris	BAArcht	Senior Engr	1.25	30	31.25	Architectural	-----	Rockwell	Proj Engr
CO Romine	BSEE	Supv, Engr'g Services	4	10	14	Engr'g Admin	I&C	Miles Labs	Mgr
✓†ER Rybarski	BSCE	Principal Engr	8.75	25	33.75	Civil	-----	Westinghse	Sr Engr
JR Smith	None	Senior Designer	1.25	14	15.25	Design/Drafting	-----	Rockwell	Designer
✓ HW Stivers	BSCE	Supv, Civil & Facs Engr'g	8.25	27	35.25	Civil	Structural	Westinghse	Sr Engr
✓†WK Stockdale	BSCE,MSCE,PhD	Manager, Civil Engr'g & Svcs	3.5	27	30.5	Struct. Dynamics	Civil/Soil Dynamcs	US Army	Colonel
†HE Wellsfry	BSCE	Senior Engr	3.5	15	18.5	Civil	Structural	Cty/Fremont	Engr
CL Whitcomb	BSEduc	Sr Engr'g Records Analyst	5.5	4	9.5	Documentation	Engr'g Admin	-----	-----

✓ Registered P.E., Washington.
† Registered P.E., other state(s).

INFORMATION ON ENGINEERS IN ENGINEERING DIVISION

DEPARTMENT: Electrical/I&C Engineering (42700)

AS OF: 7-31-81

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS	OTHER	TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
RL Abbott	BSEE	Senior Engr	2.25	12	14.25	Elect Controls	Equipt Qualif	B&W	Sr Engr
RD Bass	BSEE	Engr I	.5	6	6.5	I&C Engr	-----	Westinghse	Adv Engr
† M Basu	BSEE, MSEE	Senior Engr	1.25	6	7.25	Systems Engr'g	Equipt Qualif	Ebasco	Engr
†SR Basu	BSEE, MSEE	Supv, Electrical Engr'g	2.25	15	17.25	Elect Systems	Licensing	Bechtel/B&R	Supv
EF Boler	AAElectron	Engr II	1.5	10	11.5	Simulators	Electron Hardware	UW	Inst Tech
MR Compton	BSPhys	Senior Engr	5.25	9	14.25	Computers	Simulators	Battelle	Dvlpt Engr
✓ AJ Ebens	BSME, MSME	Program Mgmt Spec	6.75	10.25	17	Program Mgmt	Contract Admin	Westinghse	Proj Engr
WE Faraone	BSEE	Engr I	4.75	6	10.75	Equipt Qualif	Elect Field Engr'g	Bechtel	Fld Engr
†RE Green	None	Senior Engr	5.25	11	16.25	I&C Engr'g	Systems Engr'g	GE	Engr
FR Guyer	BSEE	Consulting Engr	7	18	25	Simulators	Electrical	Westinghse	Sr Engr
†AN Joshi	BSEE	Senior Engr	2.25	20	22.25	I&C Systems	Equipt Qualif	Bechtel	Sr Engr
JD Lodge	BSEE, MSNE	Senior Engr	1.5	10	11.5	Simulators	Electrical	Westinghse	Sim Engr
JT Person	BSEE	Engr II	0	2	2	Elect Distrb.	-----	EG&G	Engr
✓ JE Rhoads	BSEE	Supv, Equipt Qualif Engr'g	5	3.5	8.5	Equipt Qualif	Electrical	Bechtel	Fld Engr
RC Ruckdeschel	BSEE	Senior Engr	0	18	18	Systems	-----	Westinghse	Sr Engr
† U Shah	BSEE, MSInstE	Manager, Elect/I&C Engr'g	4.25	13	17.25	I&C	Sfty/Reliblty	NucSvc/3M	Sr Const.

✓ Registered P.E., Washington.
† Registered P.E., other state(s).

INFORMATION ON ENGINEERS IN ENGINEERING DIVISION

DEPARTMENT: Elect/I&C Engr'g - CONTINUED (42700)

AS OF: 7-31-81

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS	OTHER	TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
JP Starr	None	Engr I	5.75	20	25.75	Simulators	Electrical	Westinghse	Engr
JL Sullivan	BSEE	Senior Engr	1.25	20	21.25	Equipt Qualif	Reliability Engr'g	Hoffman	Lead Engr
✓ DT Thonn	BSEE	Principal Engr	8.5	22	30.5	Elect Systems	Protective Relay	ARCHO/GE	Spec
CJ Zeamer	BSNETech	Engr I	.5	20	20.5	Equipt Qualif	Maintenance/Elect	B&R	Engr

✓ Registered P.E., Washington.
† Registered P.E., other state(s).

INFORMATION ON ENGINEERS IN ENGINEERING DIVISION

DEPARTMENT: Nuclear Engineering (42800) AS OF: 7-31-81

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS	OTHER	TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
†SM Berry	BSME	Engr I	1.25	10	11.25	Systems Engr'g	Fuel Handling	Rockwell	Sr Engr
DE Bush	BSChE,MSNE,PhD	Principal Engr	4.75	12	16.75	Thermal Hydraulics	Containment Perf.	Stone/Wbstr	Sr Engr
†LC Chan	BSME	Engr I	0	7.5	7.5	Mechanical Systems	ThermHydr Analysis	CT Main	Mech Engr
†KR Conn	BSAE, MSAE	Engr I	4.25	19.5	23.75	Systems Analysis	Heat Balance	Gilbert Asc	Sr Engr
†GL Gelhaus	BSEPhys, MSNE	Manager, Nuclear Engr'g	7.25	7	14.25	Safety Analysis	Fuel Mgmt	TVA/BNW	Nuclear Engr
RL Heid	BSME	Principal Engr	7.5	13	20.5	Syst Perf Analysis	Component Design	Westinghse	Sr Engr
TW Horning	BSBE, MSNE	Engr I	0	5	5	Program Mgmt	Sfty/Environ Mntrg	Westinghse	Mgr,CS Anly
TH Keheley	BSME	Engr I	0	10	10	Heat Transfer	-----	Ecodyne	R&D Engr
WH Kelso	BSME	Program Mgmt Spec	3.75	17	20.75	Fuel Handling	-----	Rockwell	Sr Engr
DB Kreig	BSME	Associate Engr	0	2	2	-----	-----	NM Rsch Fac	Rsch Asst
KS Lund	BSNE	Associate Engr	.25	.75	1	-----	-----	PS NavShpyd	Eng Tech
†FJ Markowski	BSME, MSME	Senior Engr	1.75	15	16.75	Reliability	Plt Transt Analys	Westinghse	Sr Engr
TB McCall	BSEE,MSNE, PhD	Program Mgmt Spec	.25	18	18.25	Plt Analy/Design	-----	Westinghse	Prin Engr
DM Myers	BSEngPhy, MSNE	Supv, Systems & Reliablty	0	10	10	Syst Analy/Design	-----	Rogers/Asc	Sr Mgr
BD Ngo	BSME	Associate Engr	.25	0	.25	Mechanical	-----	-----	-----
RJ Nicklas	BSChE, MBA	Supv, Nuclear/Chem Engr'g	7.75	12.5	20.25	Nuclear/Chemical	Systems Engr'g	WI-Mich Pwr	Engr

✓ Registered P.E., Washington.
† Registered P.E., other state(s).

INFORMATION ON ENGINEERS IN ENGINEERING DIVISION

DEPARTMENT: Nuclear Engr'g - CONTINUED (42800) AS OF: 7-31-81

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS	OTHER	TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
FE Owen	BSChE	Senior Engr	8.25	25	33.25	Radiation Protect.	-----	UNI	Sr HP
WW Roberts	BSChE	Engr I	5.25	8	13.25	Corrosion	-----	ARCHO	Sr Engr
MR Steelman	BSEcon, BSAeroE	Engr I	3.5	2	5.5	Computer Engr'g	-----	-----	-----
RT Steen	BSME	Engr I	3.5	5	8.5	Radwaste	Systems Engr'g	Cmwlth Edisn	Gen Engr
JP Thorpe	BSNE	Engr I	1	4	5	Shielding	Fuels Cycles	NN Shpbldg	Engr
WW Waddel	BSEPhys, MSNE	Supv, Tech Programs	7	13	20	Program Mgmt	-----		

✓ Registered P.E., Washington.
† Registered P.E., other state(s).

INFORMATION ON ENGINEERS IN TECHNICAL DIV

DEPARTMENT: FUELS & ENVIRONMENT DIVISION

AS OF: AUGUST 5, 1981

Name	Degree(s)	WPPSS Position Title	Experience WPPSS	Experience Other	Experience Total	Principal Area of Expertise	Secondary Area of Expertise	Previous Employer	Title
GF Bailey (✓)	BS Chemical Engineering BS Business MS Nuclear Engineering	Manager, Fuels & Environment Division	7	16	23	Nuclear Engineering/ Management	Energy Alternatives/Environmental Engineering	Westinghouse Hanford	Manager, Environmental Engineering
RG Spencer	--	Technical Division Administrative Specialist	7	26½	33½	Financial	Administrative	UNC	Manager, Cost and Property

✓ Registered PE in Washington

Name	Degree(s)	WPPSS Position Title	Experience WPPSS	Experience Other	Experience Total	Area of Expertise	Secondary Area of Expertise	Previous Employer	Title
JR Worden	BS/MS Physics	Manager, Fuel Supply	7	17	24	Fuel Management	Nuclear Analysis	West/BNW/GE	Fuel Project Engineer
BK Boyd	BS Met Engr MS Engr Mgt	Supervisor, Fuel Services	4½	17	21½	Fuel Contract Admin. (Tech)	Fuel Quality Assurance	Argonne/ Texas Instruments/ GE	Procurement Engineer
LS Pace	BS Mathematics	Fuel Cost Engineer	5	--	5	Fuel Cost	Project Planning and Scheduling	--	--
JE Sammis	BS Ceramic Engineer	Fuel Project Engineer	7	6	13	Fuel Contract Admin (Tech)	Fuel Development	West.Hanf./ BNW	Fuel and Core Component Procurement Engineer
WG Jolly	BA Math MA Econ	Economic Evaluations Engineer	½	5	5½	Economic Analysis	Budget Forecast-	West.Hanf.	Economic Studies Engineer
JR Young (√,†)	BS CE SM CE MBA	Principal Special Studies Engineer	4	29	33	Engineer Studies/ Management	Cost Benefit Analysis/Environmental Assessment/ Economic Analysis	BNW/DUN/GE	Staff Engineer

√ Registered PE in Washington
† Registered PE in other state(s)

INFORMATION ON ENGINEERS IN TECHNICAL DIV

DEPARTMENT: ENVIRONMENTAL PROGRAMS DEPARTMENT

AS OF: AUGUST 5, 1981

Name	Degree(s)	WPPSS Position Title	Experience			Principal Area of Expertise	Secondary Area of Expertise	Previous Employer	Title
			WPPSS	Other	Total				
KA McGinnis	BS Ag Econ MS Ag Econ	Economics Evaluations Engineer	1½	4	5.	Economics	Engineer Economics/Socioeconomics	BNW	Economist
KR Wise (✓)	BS Mechanical Engineering MS Mechanical Engineering	Supervisor, Environmental Engineering	8	7	15	Supervision	Environmental Engineering	Westinghouse Hanford	Development Engineer
W Davis, III	BS Biology MS Ecology MS Management Science	Supervisor, Environmental Science	2	11	13	Ecology	Statistics	Yankee Atomic Electric Company	Environmental Scientist
RA Chitwood	BA Physics Chemistry Mathematics	Manager, Environmental Programs	10	15	25	Environmental/Regulatory Management	Siting/Licensing Management	DUN	Senior Engineer
JP Chasse (†)	BS CE MS CE-Water Resources Engineering	Senior Environmental Engineer	4	7+	11	Environmental Engineering		US EPA	Civil/Environmental Engineer
AC Rutz	BS Biology MPH-Water Quality Environmental Chemistry	Environmental Engineer	2	8	10	Environmental Engineering	Water Quality/Environemntal Chemistry	Klamath County, OR	Director of Environmental Health

✓ Registered PE in Washington
† Registered PE other state(s)

INFORMATION ON ENGINEERS IN TECHNICAL DIVISION

DEPARTMENT: ENVIRONMENTAL PROGRAMS DEPARTMENT

AS OF: AUGUST 5, 1981

Name	Degree(s)	WPPSS Position Title	Experience WPPSS	Experience Other	Experience Total	Principal Area of Expertise	Secondary Area of Expertise	Previous Employer	Title
MK Thompson	BS Geology Grad Study in Hydrology	Environmental Engineer	1½	3 res. 4 ind. & gov't	8½	Hydrology Geology	Permitting and Regulatory Compliance	Utah Dept. of Natural Resources	Engineering Geologist/ Senior Hydrologist
C Van Hoff	--	Senior State Liaison Specialist	4	5	9	Regulatory Interface	Legislative Reviewer	State of Idaho	Legislative Assistant
JC Kutt	BS Chemistry	Environmental Scientist	3 mo	10	10	Chemistry	Environmental Scientist	Battele NW Laboratory	Scientist
GS Jeane	Minor Chemistry BS Fisheries	Environmental Engineer - Biologist	5	6½	11½	Mitigation Fisheries Pollution	Water Quality	Washington State Department of Ecology	Environmentalist
WA Kiel	BS Geology BS Oceanography	Geologist	5½			Environmental Geology			
AM Lee	MUP Urban Planning; BA Urban and Regional Planning	Socioeconomic Coordinator	3	1	4	Socioeconomic Monitoring Mitigation	Land Use (physical planning)	Washington State Office of Community Development	Administrative Intern
ML Miller	BS Physical Science MS Health Physics	Environmental Scientist	1½	3	4	Health Physics	Computer Programmer	Exxon	

INFORMATION ON ENGINEERS IN TECHNICAL DIVISION

DEPARTMENT: ENVIRONMENTAL PROGRAMS DEPARTMENT

AS OF: AUGUST 5, 1981

Name	Degree(s)	WPPSS Position Title	Experience WPPSS	Experience Other	Experience Total	Principal Area of Expertise	Secondary Area of Expertise	Previous Employer	Title
JE Mudge	PhD MS MEd BS	Senior Environmental Scientist	3	8	11	Aquatic Ecology	Physiology Bioassay	Metropolitan Edison Company/ Texas Instrument Inc.	Supervisor, Radiation Safety and Environmental Engineer/Aquatic Ecology Task Leader
FD Quinn	BS Meteorology 2+ Yrs Grad School	Senior Environmental Scientist	1	20	21	Air Pollution Meteorology	Scientific Systems Analyst	USAF/Air Wea. Svc.	Technical Division Meteorological Systems Analyst
LS Schleder	AA Zoology, Environmental Studies	Environmental Technician	2 (8mos Security Guard for WPPSS)	11	13	Environmental Technician	Security Guard	Industrial Fiberglass/ Shelbies Fiberglass	Miscellaneous Fabrication

INFORMATION ON ENGINEERS IN ENGINEERING DIVISION

DEPARTMENT: WNP-2 ENGINEERING
62900

AS OF: 8-6-81

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS	OTHER	TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
HL Bennett	BSME	Senior Engineer	5.25	21	37	HVAC	Valves	Industrial Pipe/Supply	Equipt Engr
+✓ HA Crisp	BSME, MSPE, MSCE	Project Engineering Management Specialist	1	25	26	Civil/Structural/ Management	Systems	U. S. Navy	Captain/Navy
+✓ EA Fredenburg	BSCE, MBA	Principal Engineer	5.25	6	11.25	Structural	Cont.Des/Prog Mgmt	Bechtel	Sr. Engr.
PW Harness	BSME	Manager, Field Engineering (Acting)	8.75	5	13.75	Systems Engrg.	Rotating Equipm't.	UNC	Plant Engineer
BA Holmberg	BS Marine Engr.	Deputy Project Manager-Engrg.	2.75	20	22.75	Management/Nuclear/ Operations	Mechanical	U.S. Navy	Commander/Navy
+ JH Ralphs	BS Engineering MS Gen. Engrg.	Senior Engineer	3.5	25	28.5	Mechanical	Chemical	Bechtel	Sr. Field Engr
TA Stanley	BSME	Project Engrg. Mgmt Spec.	4.25	8.5	12.75	Systems	Construction Engrg.	ITT	Project Engr.
DC Timmins	BS Physics	Project Engrg Mgmt Spec	5.3	7.7	13	Nuclear/Mech.	Chemical Engrg/ Shielding	Bechtel	Supervisor Nuclear Analys
✓ LW Vance	BSCE	Project Engrg Mgmt Spec	6	10	16	Civil	Highway	Wash. State Dept. Transportation	Sr. Engineer
✓ MF Wiitala	BSEE	Principal Engineer	9	25	34	Electrical	---	Westinghouse	Sr. Engineer

✓ Registered P. E., Washington
\+ Registered P. E., Other State(s)

INFORMATION ON HEALTH, SAFETY & SECURITY MANAGEMENT AND STAFF

DEPARTMENT: HEALTH, SAFETY & SECURITY

AS OF: August 1981

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
J.W. Shannon	BS Physics (equivalency)	Manager, Health, Safety & Security	7 yr./23 yr./30 yr.	Construction Startup Operations	Chemical Processing Operational Testing Radiation Protection Emergency Preparedness	Dairyland Power Cooperative	Supervisor, Health & Safety Dept.

+Registered P. E., Washington.
✓Registered P. E., other state(s).

INFORMATION ON HEALTH, SAFETY & SECURITY MANAGEMENT AND STAFF

DEPARTMENT: INDUSTRIAL SAFETY & FIRE PROTECTION

AS OF: August 1981

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
W.E. Taylor	--	Manager, Industrial Safety & Fire Protection	1/21/22 yrs.	Fire & Radiation Protection Health & Safety Programs Fire & Radiation Testing	Laboratory Operations Emergency Preparedness Reactor Testing	Westinghouse Hanford	Manager, Health & Safety Group
R.M. Gough	AD Fire Protection BS Trade & Industrial	Supervisor, Loss Control Services	7/13/20 yrs.	Fire Protection/ Industrial Safety	Firefighting Nuclear Operations, Research, & Construction	Westinghouse	Industrial Safety & Fire Prot. Specialist
J.C. Bell	BS Mathematics	Supervisor, WNP-2 Safety	12 yrs.	Nuclear Operations Construction, & Research	Turbine Manufacturing Utility Market Planning	Westinghouse Electric	Generation Engineer
M.J. Hoylman	--	Specialist I	4 mo./11/11 yrs.	Occupational Health Testing Emergency Medicine Industrial Safety	Power Plant Construction Personnel Labor Relations	Kaiser Engineers	Safety Supervisor
D.T. Evans	BS Mechanical Engineering	Senior Engineer-Fire Protection	2 mo./9/9 yrs.	Design	Fire Protection Systems Evaluation	Western Div. Naval Facilities Engineering	Assistant Branch Head
G.E. Towne	US Air Force	Senior Specialist	3/29/32 yrs.	Developed & implemented Hanford Fire Training Manual, Supply System Fire Brigade Training Programs & Manuals	Developed & implemented Hanford Emergency Reserve Procedures, Radiation Casualty Handling for DSHS	Rockwell Hanford Co.	Manager, Fire Equip. Services

+ Registered P. E., Washington.

✓ Registered P. E., other state(s).

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DEPARTMENT: INDUSTRIAL SAFETY & FIRE PROTECTION

AS OF: August 1981

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
K.C. Bleiler	--	Training Specialist I	1 mo./22/22 yrs.	Emergency Medical & Fire Protection	Emergency Medical Technician Certified Instructor in First Aid	Columbia Oil Co.	Vending Supervisor
D.G. Parthree	--	Supervisor, WNP-1/4 Safety & Fire Protection	1 mo./19/19 yrs.	Nuclear Construction, Startup & Operation	Industrial Safety Fire Protection Emergency Preparedness	Gibbs & Hill/ D.U.C.I.	Manager, Loss Prevention
R.A. Fisher	--	Specialist I	1/21/22 yrs.	Industrial Fireman Nucl./Chem. Operator	Fire Detection Testing Equipment & Design for Fire Protection	Interlakes School	Custodian
D.A. Smith	BA Psychology	Supervisor, WNP-3/5 Safety	3/5/8 yrs.	Power Plant Construction Safety	Fossile Fuel Construction Safety	WSH/Boecon/ GERI, WNP-2	Safety Mgr.

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INFORMATION ON HEALTH, SAFETY & SECURITY MANAGEMENT AND STAFF

DEPARTMENT: WNP-2 HEALTH PHYSICS/CHEMISTRY

AS OF: August 1981

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
R.G. Graybeal	BS Education	Manager, Health Physics/ Chemistry, WNP-2	6/21/27 yrs.	Construction, Startup & Operations	Industrial Safety Programs	Iowa Electric Light & Power	Radiation Protection Engineer
L.D. Morrison	BS Biochemistry	Chemistry Supervisor	2/2/4 yrs.	Laboratory Analysis	Plant Chemistry Control	US Testing Co.	Research Scientist
L.G. Berry	--	Health Physics Supv.	2 mo./12/12 yrs.	Operation & Shut-down Activity Surveys	Plant Radiation Surveillance Health Physics Program	Lawrence Livermore Lab.	Health & Safety Technician

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DEPARTMENT: WNP-1/4 HEALTH PHYSICS/CHEMISTRY AS OF: August 1981

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
V.E. Shockley	BS Geology	Manager, Health Physics/ Chemistry, WNP-1/4	2 yrs./20/22 yrs.	Radiological & Safety Hazards Analysis	Respiratory & Contamination Monitoring	Consumers Power Co.	Plant Health Physicist
J.D. Mills	--	Health Physics Supv.	1/22/23 yrs.	Fuel, Air & Radiation Analysis Nuclear Rockets Testing	Radiation & Contamination Surveys	Consumers Power Co.	Supv., Radiation Protection
R.M. Brun	US Navy Nuclear Training School	Chemistry Supv.	2/16/18 yrs.	Mechanical Operator Engineering Lab. Technician	Chemistry & Radiochemistry Analysis Plant Policy Administration	Consumers Power Co.	Supv., Chemical & Radiation Protection

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DEPARTMENT: SECURITY

AS OF: August 1981

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
S.R. Telander	USAF Nuclear Weapons, Security Police Officer Academy, Missile Staff Officer	Manager, Security Dept.	4/20/24 yrs.	Operational Physical Security Management	--	USAF	Nuc. Security & Law Enforcement Officer
S.H. Martin	BS Business Mgmt.	Security Force Chief	3/26/29 yrs.	Operational Physical Security Management	--	USAF	Deputy Chief of Security Police
D.M. Vorheis	AA Business Mgmt.	Security Force Captain	3/23/26 yrs.	Operational Physical Security Management	Systems & Equipment Specifications	USAF	Chief, Security Police
R.J. Marzano	BA Police Science	Security Force Captain	3/24/27 yrs.	Operational Physical Management	Security Force Technical Training	USAF	Colonel, Security Police
J.W. Klingelhoefer, Jr.	BS Engineering	Administrative Security Supervisor	2/7/9 yrs.	Nuclear Security	Army Artillery	NUSAC, Inc.	Sr. Technical Associate
R.J. Givin	BS Law Enforcement MS Criminal Justice	Supv., Security Training & Evaluation	3/6/9 yrs.	Training Systems Engineer, Installation Physical Security Officer	Security Force Training	US Military Police	Curriculum Officer

+ Registered P.E., Washington

✓ Registered P.E., other state(s).

INFORMATION ON HEALTH, SAFETY & SECURITY MANAGEMENT AND STAFF

DEPARTMENT: RADIOLOGICAL PROGRAMS

AS OF: August 1981

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
D.E. Larson	US Navy Nuclear Power School College equivalency	Manager, Radiological Programs	7/13/20 yrs.	Health Physics Radiological Protection	Engineering Lab. Technician	LaSalle Boiling Water Reactor	Sr. Health Physics Technician
J.V. Everett	BS Computer Science/Physics MS Nuclear Engineering/Health Physics	Supv., Emergency Preparedness	3/2/5 yrs.	Emergency Planning	Radiological Engineer	Vallecitos Nuclear Center	Radiological Engineer
J.R. Allen	BS Physics ME Nuclear Engineering/Health Physics	Emergency Planning Specialist	1/3/4 yrs.	Emergency Plan & Procedure preparation	Radiation Specialist	USNRC	Radiation Specialist
D.H. Oatley	BS Biology MS Health Physics	Emergency Planning Specialist II	1/4/5 yrs.	Health Physics	Emergency Preparedness Planning Radiological Engineering Design	Portland General Electric	Health Physics Engineer
A.F. Klauss	--	Emergency Planning Specialist	3/9/12 yrs.	Law Enforcement	Emergency Planning	PA State Police	Trooper
D.H. Mannion	BS Business Administration MS Management	Emergency Planning Specialist	4 mo./10/10 yrs.	Nuclear Weapons Programs	Nuclear Weapons Operations	US Army	Executive Officer
R.F. Haight	BS Physics MS Physics	Supv., Dose Assessment & Audit Section	3/7/10 yrs.	Health Physics	Pathway Modeling, Dose Assessment	UNC Nuclear	Sr. Health Physicist

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✓ Registered P. E., other state(s).

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DEPARTMENT: RADIOLOGICAL PROGRAMS

AS OF: August 1981

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
G.V. Oldfield	BA Physics MPH	Health Physicist	2/7/9 yrs.	Emergency Response & Preparedness	ALARA Design Review	UNC Nuclear Industries	Sr. Health Physicist
D.B. Ottley	BS Physics	Health Physicist	2/4/6 yrs.	Radiation Control	Risk Assessment Emergency Preparedness	UNC Nuclear	Safety Engineer
J.O. Parry	BS Applied Physics	Health Physicist	1/6/7 yrs.	Health Physics	ALARA Review	Commonwealth Edison	Supv. Rad. Protection Chemistry
R.W. Craig	College equivalency Health Physics	Supv., Radiological Programs	4/18/22 yrs.	Construction Startup Operations	Critical Facility/ Fuel Fabrication	Dairyland Power Cooperative	H/P Technician
C.A. Graybeal	AA	Dosimetry Analyst	1/2/3 yrs.	Recording Radiation Exposure	Respiratory Protection Records	Richland School District	Secretary
J.D. Artis III	AA Envir. HP/C BS Envir. Science Grad Studies	Dosimetry Analyst	1/6/7 yrs.	Chemical Testing of Water & Waste Water	External Dosimetry Measurement Program	USAF	Sergeant
T.J. Froelich	BS Physics MS Envir. Health Science	Health Physicist I	1/7/8 yrs.	Health Physics	Power Plant Construction	Battelle	Research Scientist
Y.E. Derrer	--	Training Spec. I	5 mo./11/11 yrs.	Health Physics	Radiological Medical Fields	Public Service Co. of Colorado	Health Physics Technician
R.C. McGillic	US Navy Nuclear Power School	Training Spec. I	1½/10/11½ yrs.	Engineer Officer of the Watch	Training, Health Physics	Health Physics Systems, Inc.	Sr. Health Physicist

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✓ Registered P. E., other state(s).

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DEPARTMENT: RADIOLOGICAL PROGRAMS AS OF: August 1981

NAME	DEGREE(S)	WPPSS POSITION TITLE	EXPERIENCE WPPSS/OTHER/TOTAL	PRINCIPAL AREA OF EXPERTISE	SECONDARY AREA OF EXPERTISE	PREVIOUS EMPLOYER	TITLE
G.D. Rhinehart, Jr.	US Navy Nuclear Power School	Training Spec. I	1/10/11 yrs.	Operations Radiation Protection	Safety Fire Protection	Omaha Public Power District	H.P. Technician
R.E. Broz	BS Physiology	Training Spec. I	1 mo./10/10 yrs.	Health Physics	Emergency Preparedness	State of WA	Health Physicist

+ Registered P. E., Washington.
√ Registered P. E., other state(s).

Q. 040.075

Provide a table that lists all equipment including instrumentation and vital support systems equipment required to achieve and maintain hot and/or cold shutdown. For each equipment listed:

a. Differentiate between equipment required to achieve and maintain hot shutdown and equipment required to achieve and maintain cold shutdown;

b. Define each equipment's location by fire area;

c. Define each equipment's redundant counterpart;

d. Identify each equipment's essential cabling (instrumentation, control and power). For each cable identified: (1) describe the cable routing (by fire area) from source to termination, and (2) identify each fire area location where the cables are separated by less than a wall having a three-hour fire rating from cables for any redundant shutdown system; and,

e. List any problem areas identfied by d.2 above that will be corrected in accordance with Section III.G.3 of Appendix R (i.e., alternate or dedicated shutdown capability).

Response

The method utilized to assure that all equipment, including instrumentation and vital support system equipment required to achieve and maintain hot and/or cold shutdown, remains available during a fire is described below.

The plant was divided into distinct fire areas; the boundary walls, floor and ceiling of which are capable of containing any fire internal to the area. Each fire area is then reviewed and analyzed individually.

Two methods of analysis are available; the particular analytical method chosen depends upon the nature of the fire area. For the case where redundant systems required to achieve and maintain hot and/or cold shutdown are located in independent fire areas, a Redundant Area concept was utilized for analysis. For the case where redundant systems required to achieve and maintain hot and/or cold shutdown are located in the same fire area, a Dedicated Shutdown concept was utilized.

A dedicated flowpath consisting of a list of systems required to shutdown the plant was developed. Criteria included a loss of offsite power, no single failure (other than a single fire and its effects), and that manual operation after 72 hours was acceptable.

To show the redundant equipment and differentiate between equipment required to achieve and maintain hot shutdown and equipment required to achieve and maintain cold shutdown, Table 040.075 is divided into groups of systems, categorized by their functions, as described below.

040.075.1a,b	RHR System
040.075.2a,b	SW System
040.075.3a,b	RCIC System
040.075.4a,b	Nuclear Boiler System
040.075.5	Miscellaneous Systems
040.075.6	HVAC Systems

Instruments and process components including pumps, heat exchangers, fans and remote operated valves are designated in the above tables for each of the modes of reactor shutdown, as follows:

Symbol	Definition
VC	Remote operated valve is closed and serves as the pressure boundary for the dedicated flow path pertaining to this mode.
VO	Remote operated valve is open as part of the dedicated flow path for this mode.
EO	Equipment other than valves which is required to be operational in this mode.
NR	Remote operated valve, instrument or equipment not required for this mode.
IO	Instrument is required to be operational for this mode.

The following illustrates specific examples from Table 040.075.1:

(1) Motor operated valve RHR-V-24B is required to be open for mode A-1 only (event to hot shutdown).

(2) Motor operated valve RHR-V-42B is required to be open for mode A-2 only (hot shutdown to cold shutdown).

(3) Motor operated valve RHR-V-3B is required to be open for both modes A-1 and A-2.

(4) Motor operated valve RHR-V-16B serves as a pressure boundary and must remain closed during both operating modes.

The table also lists the fire zones for each dedicated piece of equipment as well as tabulating its redundant counterpart.

Conduit and tray layout working drawings were marked up to show the boundaries of each fire area. Areas were identified on the drawings with a fire area designation tag that was unique for each area. Based on a preliminary review of the area, the drawings were marked up to identify it as a Divisional area or a General area. Divisional areas were those areas where review indicated that virtually all equipment in the area was assigned to one of the three major safety-related electrical separation divisions (Divisions 1, 2 or 3). The room was then assigned to that division, and analysis proceeded utilizing the Redundant Area method. General areas were those areas where review indicated that there was significant mixing of equipment assigned to more than one of the three major safety-related electrical separation divisions. Analysis for these areas utilized the Dedicated Shutdown method.

Areas analyzed via the Redundant Area method were reviewed, and raceway working drawings marked up to identify all cables and equipment located therein that are not compatible with (intrude upon) the electrical division to which the room has been assigned. Intruding equipment and cables are then analyzed to verify that the effects of fire upon them do not defeat the overall capability of the safety-related systems of the intruding divisions to achieve safe shutdown. If the plant can be safely shut down, then no fix is required. If the plant cannot be safely shut down, then the intruding cables and/or equipment must be protected in accordance with the guidelines of Appendix R or relocated to another fire area. Any relocated equipment must then be included in the analysis for the fire area into which it was relocated. This method takes advantage of the separation of redundant equipment into different fire areas that is an inherent part of plant layout and design.

Areas analyzed via the Dedicated Shutdown method were reviewed, and raceway drawings marked up, to identify all cables and equipment located therein which are part of the Dedicated Shutdown System. The Dedicated Shutdown method involved selecting systems necessary to achieve and maintain hot and/or cold shutdown, and protecting minimum specified equipment within those systems to insure that these dedicated systems are available for safe shutdown. Any dedicated shutdown equipment, and dedicated shutdown cables that are required to be functional, identified in an area analyzed via this method must be protected in accordance with the guidelines of Appendix R or relocated to a divisional fire area. Any relocated equipment must then be included in the analysis for the fire area into which it was relocated.

a. For areas analyzed via the Redundant area method, a table listing specific equipment required to achieve and maintain hot and/or cold shutdown is not appropriate. Basic plant design dictates that shutdown can be achieved utilizing equipment of only two of the three major electrical separation divisions. The intent of redundant analysis is not to show that a minimum amount of equipment remains available to achieve shutdown. Rather, it takes no credit for operation of equipment of the division to which the fire area is assigned, and verifies that loss of equipment in the other two divisions does not occur to the extent that their overall capability to provide for safe shutdown is defeated. Table 040.075.9 lists the plant systems contained in the safety-related electrical separation divisions.

   Table 040.075.7 and 040.075.8 list all electrical equipment and cables that are a part of the Dedicated Shutdown System.

   Differentiation between equipment needed to achieve and maintain hot and/or cold shutdown is not provided for the Redundant Area method of analysis, since there is no specific equipment listing. Tables 040.075.1 thru 6 provide this differentiation for mechanical/nuclear equipment of the Dedicated Shutdown System. Since virtually all dedicated electrical equipment supports both hot and cold shutdown, no distinction is made.

   Tables 040.075.1 thru 6 list the fire areas of all dedicated shutdown equipment except electrical distribution equipment.

b. Table 040.075.7 indicates the fire area location of all safety-related electrical equipment in the plant. Table 040.075.10 indicates the physical location of all fire areas. The equipment that is a part of the Dedicated Shutdown System is so noted.

c. Tables 040.075.1 thru 6 indicate redundant counterparts for Dedicated Shutdown System equipment, except for electrical distribution system equipment.

Discussion of redundant electrical equipment counterparts can only be discussed in general terms. This applies to both the Redundant Area and Dedicated Shutdown concepts. The overall plant design provides for redundancy of the major safety-related electrical divisions as follows:

Division 1: Redundant counterpart is the Divisions 2 and 3 combination.

Division 2: Redundant counterpart is Division 1.

Division 3: Redundant counterpart is Division 1.

At the electrical distribution system source end (standby ac diesel-generators and standby dc batteries), identification of directly redundant counterparts are relatively straight-forward. However, downstream in the distribution system, a specific discussion of electrically redundant counterparts becomes less meaningful, particularly in the two major electrical divisions (Divisions 1 and 2). This occurs because the duality existing in the electrical distribution system equipment layout does not necessarily follow through to the connection of loads (pumps, fans, valves, etc.) to this equipment. For this reason, indication of specific electrical distribution system redundant counterparts is not provided.

d. Specific identification of essential cabling is not required in connection with the Redundant Area method, due to the analytical concept utilized. Table 040.075.8 provides identification of power, control and instrumentation cabling required for operation of Dedicated Shutdown System equipment. The following should be noted:

1. Division 1 dc supplied RCIC equipment has power train cabling tracked back only as far as the battery. Battery capacity is sufficient to meet system operational needs.

2. Division 1 ac supplied valves have power train cabling tracked back only as far as the immediate supply source (mcc). These valves are not required to change position for shutdown.

3. In the interest of providing a complete listing of cables and facilitating a manageable study, all power, control and instrumentation cabling directly required for operation of Dedicated Shutdown equipment has been included in the cable listing. These cables have also been marked on raceway drawings and identified in the dedicated fire area analysis. However, there are cases where protection of cabling is not, in fact, required.

   For example, a three phase MOV that is open for power operation, hot shutdown and cold shutdown (See Table 040.075.1 thru 6) does not require protection of power cabling. Detailed analysis of Dedicated Fire Areas will identify cases such as this.

Intruding and Dedicated Shutdown cabling and equipment for which analysis indicates protection is required will be either:

1. Relocated to another (acceptable) fire area.

2. Separated or protected in accordance with Appendix R, Section III.G.

e. Complete details of analysis, identification of problem areas, and description of corrective measures will be provided in the Fire Hazards Shutdown Analysis to be included in Appendix F.

TABLE 40.75.1a

EQUIPMENT LIST - DEDICATED FLOW PATH
RESIDUAL HEAT REMOVAL SYSTEM

(B&R Dwg. M-521)

Tag No.	Description	Event to Hot Shutdown (Mode A-1)	Hot Shutdown to Cold Shut-down (Mode A-2)	Fire Area of Dedicated Component	Redundant Counterpart
RHR-HX-1B	RHR Heat Exchanger	EO	EO	R-IV	RHR-HX-1A
RHR-P-2B	RHR Pump "B"	EO	EO	R-IV	RHR-P-2A
RHR-V-3B	HX Outlet	VO	VO	R-IV	RHR-V-3A
RHR-V-4B	Suppression Pool Water Supply	VO	VO	R-IV	RHR-V-4A
RHR-V-6B	Shutdown Water Supply	VC	VC	R-IV	RHR-V-6A
RHR-V-11B	Isolation	VC	VC	R-XVII	RHR-V-11A
RHR-V-16B	Drywell Spray Isolation	VC	VC	R-I	RHR-V-16A
RHR-V-23	RCIC Head Spray	VC	VC	R-I	NONE
RHR-V-24B	Test Line Isolation, Loop B	VO	VC	R-XVII	RHR-V-24A
RHR-V-26B	Condensate Return to RCIC	VC	VC	R-XVII	RHR-V-26A
RHR-V-27B	Suppression Spray Isolation	VC	VC	R-XVII	RHR-V-27A
RHR-V-42B	LPCI Isolation	VC	VO	R-I	RHR-V-42A
RHR-V-47B	HX Inlet	VO	VO	R-IV	RHR-V-47A
RHR-V-48B	HX Bypass	VC	VC	R-IV	RHR-V-48A
RHR-V-49	HX Warmup	VC	VC	R-IV	NONE
RHR-V-52B	Steam Supply from RCIC	VC	VC	R-IV	RHR-V-52A
RHR-V-53B	Shutdown Return Isolation	VC	VC	R-I	RHR-V-53A
RHR-V-60B	Sample Line Block	VC	VC	R-IV	RHR-V-60A
RHR-FCV-64B	Minimum Flow	VO	VO	R-IV	RHR-FCV-64A
RHR-V-74B	HX Vent	VC	VC	R-IV	RHR-V-74A
RHR-V-115	RHR/SW Intertie	VC	VC	R-IV	NONE

TABLE 40.75.1b

INSTRUMENT LIST - DEDICATED FLOWPATH
RESIDUAL HEAT REMOVAL SYSTEM

(B&R Dwg. M-521)

Instrument Tag No.	Description	Event to Hot Shutdown (Mode A-1)	Hot Shutdown to Cold Shut-down (Mode A-2)	Fire Area of Dedicated Component	Redundant Counterpart Tag No.
E12-FE-N014B	Orifice Plate	IO	IO	R-I	NONE
C61-FT-N001	Flow Transmitter	IO	IO	later	E12-FT-N015B
C61-FI-R005	Flow Indicator	IO	IO	RC-IX	E12-FI-R603B

040.075-8

TABLE 40.75.2a

EQUIPMENT LIST - DEDICATED FLOWPATH
SERVICE WATER SYSTEM

(B&R Dwg. M-524)

Tag No.	Description	Event to Hot Shutdown (Mode A-1)	Hot Shutdown to Cold Shut-down (Mode A-2)	Fire Area of Dedicated Component	Redundant Counterpart
SW-P-1B	Standby Service Water Pump	EO	EO	SW-II	SW-P-1A
SW-V-2B	SW-P-1B Discharge	VO	VO	SW-II	SW-V-2A
SW-V-4B	Diesel "B" Return	VO	VC	DG-III	SW-V-4A
DCW-V-6B1	Diesel "B" Heat Exchangers Cooling Supply	VO	NR	DG-III	DCW-V-6A1
DCW-V-6B2	Diesel "B" Heat Exchangers Cooling Supply	VO	NR	DG-III	DCW-V-6A2
SW-TCV-11B	Control Room Cooling	VO	VO	RC-XII	SW-TCV-11A
SW-V-12B	Spray Pond Return	VO	VC	SW-II	SW-V-12A
SW-V-24B	RHR "B" Pump Room Cooling	VO	VO	R-IV	SW-V-24A
SW-V-34	RCIC Pump Room Cooling	VO	VC	R-VI	NONE
SW-PCV-38B	Loop Pressure Control	VO	VO	SW-II	SW-PCV-38A
RHR-V-68B	RHR HX Outlet	VO	VO	R-IV	RHR-V-68A
SW-V-69B	Cooling Tower Return	VC	VO	SW-II	SW-V-69A
SW-V-75B	Fuel Pool Emergency Make up	VC	VC	R-I	SW-V-75A
SW-V-90	DG "B" Cable Cooling	VC	VC	TG-I	NONE
SW-V-187B	FPC Supply	VC	VC	R-XIV	SW-V-187A
SW-V-188B	FPC Return	VC	VC	R-XIV	SW-V-188A

TABLE 40.75.2b

INSTRUMENT LIST - DEDICATED FLOWPATH
SERVICE WATER SYSTEM

(B&R Dwg. M-524)

Tag No.	Description	Event to Hot Shutdown (Mode A-1)	Hot Shutdown to Cold Shut-down (Mode A-2)	Fire Area of Dedicated Component	Redundant Counterpart Tag No.
SW-PI-32BR	Pressure Indicator	IO	IO	RC-IX	SW-PI-32B
SW-PT-32BR	Pressure Transmitter	IO	IO	SW-II	SW-PT-32B
SW-TE-1BR	Temperature Element	IO	IO	SW-II	SW-TE-1C
SW-TI-1BR	Temperature Indicator	IO	IO	RC-IX	SW-TI-1C
SW-LTD-1BR	Level Transducer	IO	IO	SW-II	SW-LTD-1C
SW-LT-1BR	Level Transmitter	IO	IO	SW-II	SW-LT-1C
SW-LI-1BR	Level Indicator	IO	IO	RC-IX	SW-LI-1C

TABLE 40.75.3a

EQUIPMENT LIST - DEDICATED FLOWPATH
REACTOR CORE ISOLATION COOLANT SYSTEM

(B&R Dwg. M-519)

Tag No.	Description	Event to Hot Shutdown (Mode A-1)	Hot Shutdown to Cold Shut-down (Mode A-2)	Fire Area of Dedicated Component	Redundant Counterpart
RCIC-P-1	RCIC Pump	EO	NR	R-IV	NONE
RCIC-P-2	Condenser Vacuum Pump	EO	NR	R-VI	NONE
RCIC-P-4	Condenser Condensate Pump	EO	NR	R-VI	NONE
RCIC-HX-1	Barometric Condenser	EO	NR	R-VI	NONE
RCIC-TK-1	Vacuum Tank	EO	NR	R-VI	NONE
RCIC-VT-1	RCIC Turbine	EO	NR	R-VI	NONE
RCIC-V-8	Outboard Containment Isolation	VO	NR	R-I	NONE
RCIC-V-63	Inboard Containment Isolation	VO	VC	R-II	NONE
RCIC-V-64	Outboard Containment Isolation (RHR-HT-EX)	VC	VC	R-I	NONE
RCIC-V-45	Turbine Isolation	VO	NR	R-VI	NONE
RCIC-V-1	Turbine Trip	VO	NR	R-VI	NONE
RCIC-V-2	Turbine Governing	VO	NR	R-VI	NONE
RCIC-V-68	Turbine Exhaust	VO	VC	R-I	NONE
RCIC-V-69	Vacuum Pump Exhaust	VO	VC	R-VI	NONE
RCIC-V-76	Inboard Containment Isolation Valve By-pass	VC	VC	R-II	NONE
RCIC-V-10	Cond. Storage TK Supply	VC	NR	R-VI	NONE
RCIC-V-13	RCIC Injection to RPV	VO	VC	R-I	NONE
RCIC-V-19	Minimum Flow	VO	VC	R-VI	NONE
RCIC-V-22	Cond. Storage TK Return	VC	NR	R-VI	NONE
RCIC-V-31	Suppression Pool Supply	VO	VC	R-VI	NONE
RCIC-V-46	Turbine Lube Oil Cooling	VO	NR	R-VI	NONE
RCIC-V-80	Outboard Isolation Containment Vacuum	VC	VC	R-I	NONE

TABLE 40.75.3b

INSTRUMENT LIST - DEDICATED FLOWPATH
REACTOR COOLANT INJECTION SYSTEMS

(B&R Dwg. M519-RCIC)

Tag No.	Description	Event to Hot Shutdown (Mode A-1)	Hot Shutdown to Cold Shut-down (Mode A-2)	Fire Area of Dedicated Component	Redundant Counterpart Tag No.
E51-FE-N001	Flow Element	IO	IO	R-VI	NONE
E51-FT-N003	Flow Transmitter	IO	IO	R-I	NONE
C61-FIC-R001	Flow Controller	IO	IO	RC-IX	E51-FIC-R600
C61-SQRT-K001	Square Root Extractor	IO	IO	RC-IX	E51-SQRT-K601
C61-FI-R001-1	Flow Indicator	IO	IO	RC-IX	E51-FI-R601-1
C61-SI-R003	Speed Indicator	IO	IO	RC-IX	E51-C002

040.075-13

TABLE 40.75.4a

EQUIPMENT LIST - DEDICATED FLOWPATH
NUCLEAR BOILER SYSTEM

(B&R Dwg. M-529)

Tag No.	Description	Event to Hot Shutdown (Mode A-1)	Hot Shutdown to Cold Shut-down (Mode A-2)	Fire Area of Dedicated Component	Redundant Counterpart
RPV	Reactor Pressure Vessel	EO	EO	R-II	NONE
**MS-RV-4A	MS/SRV Valve	NR	VO	R-II	MS-RV-4B
**MS-RV-4C	MS/SRV Valve	NR	NR	R-II	MS-RV-3D
**MS-RV-4D	MS/SRV Valve	NR	NR	R-II	MS-RV-5C
MS-V-22A	Isolation Valve	VC	VC	R-II	NONE
MS-V-22B	Isolation Valve	VC	VC	R-II	NONE
MS-V-22C	Isolation Valve	VC	VC	R-II	NONE
MS-V-22D	Isolation Valve	VC	VC	R-II	NONE

**3 valves required for Mode A-2, i.e., for RHR "Special Shutdown Cooling"

TABLE 40.75.4b

INSTRUMENT LIST - DEDICATED FLOWPATH
NUCLEAR BOILER SYSTEM

(B&R Dwg. M-529)

Tag No.	Description	Event to Hot Shutdown (Mode A-1)	Hot Shutdown to Cold Shut-down (Mode A-2)	Fire Area of Dedicated Component	Redundant Counterpart Tag No.
B22-LIS-N037A	Level Indicating Switches	IO	IO	R-I	B22-LIS-37B
B22-LIS-N037C	Level Indicating Switches	IO	IO	R-I	B22-LIS-37D
B22-LITS-N026D	Level Transmitter	IO	IO	R-I	NONE
C61-LI-R010	Level Indicator	IO	IO	RC-IX	NONE
C61-PT-N006	Pressure Transmitter	IO	IO	R-I	B22-PT-N51A
C61-PI-R011	Pressure Indicator	IO	IO	RC-IX	B22-PR-R603

TABLE 40.75.5

INSTRUMENT LIST - DEDICATED FLOWPATH MISCELLANEOUS SYSTEMS

Tag No.	Description	Event to Hot Shutdown (Mode A-1)	Hot Shutdown to Cold Shut-down (Mode A-2)	Fire Area of Dedicated Component	Redundant Counterpart Tag No.
CMS-TE-19R, 37R,39R,41R, 42R,43R,44R	Temperature Elements	IO	IO	R-II	CMS-TE-20, 36,40 SPTM-TE-1A/1B
CM-MV/I-19R, 37R,39R,41R, 42R,43R,44R	MV to Current Converters	IO	IO	RC-IX	CMS-MV/I-20, 36,40
CMS-TI-19R, 37R,39R,41R, 42R,43R,44R	Temp. Indicators	IO	IO	RC-IX	CMS-TI-20, 36,40 SPTM-TR-1/2
CMS-PT-2R,6R	Pressure Trans.	IO	IO	R-I	CMS-PT-1, 5
CMS-LT-2R	Lower Trans.	IO	IO	R-IV	CMS-LT-1
CMS-PI-2R,6R	Press. Indicators	IO	IO	RC-IX	CMS-PR-1
CMS-LI-2R	Level Indicators	IO	IO	RC-IX	CMS-LR/PR-3

TABLE 40.75.6

EQUIPMENT LIST - DEDICATED FLOWPATH
HVAC SYSTEMS

Tag No.	Description	Event to Hot Shutdown (Mode A-1)	Hot Shutdown to Cold Shut-down (Mode A-2)	Fire Area of Dedicated Component	Redundant Counterpart
	Reactor Building				
RRA-FN-3	RHR Pump Room "B" Fan Coil Unit	EO	EO	R-IV	RRA-FN-2
RRA-FN-6	RCIC Pump Room Fan Coil Unit	EO	NR	R-VI	RRA-FN-4
RRA-FN-10	Div. II MCC Room Fan Coil Unit	EO	EO	R-XVIII	RRA-FN-11
RRA-FN-12	Div. I MCC Room Fan Coil Unit	EO	NR	R-XX	NONE
RRA-FN-14	H_2 Recombiner MCC Room Fan Coil Unit	EO	EO	R-XIX	RRA-FN-13
	Control Building				
WMA-FN-51B	Control Room Air Handling Unit	EO	EO	RC-XII	WMA-FN-51A
WMA-TIC-11B	Control For SW-TCV-11B	EO	EO	RC-XII	WMA-TIC-11A
WMA-FN-52B	Cable Spreading Room Air Handling Unit	EO	EO	RC-XII	WMA-FN-52A
WMA-AD-52-1	Return Air Damper	EO	EO	RC-XII	NONE
WMA-AD-52-2	Return Air Damper	EO	EO	RC-XII	NONE
WMA-FN-53B	SWGR Area Air Handling Unit	EO	EO	RC-XII	WMA-FN-53A
	Pump Houses				
PRA-FN-1B	SSW "B" Pump Fan Coil Unit	EO	EO	SW-II	PRA-FN-1A
	Diesel Generator Building				
	Div. II				
DMA-FN-22	DG & Elect. Rm Air Handling Unit	EO	EO	DG-III	DMA-FN-12
DMA-FN-21	DG Room Air Handling Unit	EO	EO	DG-III	DMA-FN-11
DEA-FN-22	Day Tank Room Exhaust Fan	EO	EO	DG-III	DEA-FN-12
DEA-FN-21	DG Room Exhaust Fan	EO	EO	DG-III	DEA-FN-11

TABLE 040.75-7

LOCATION OF MAJOR SAFETY RELATED

ELECTRICAL DISTRIBUTION EQUIPMENT

AREA	BLDG	EQUIPMENT	ELEV	DIV	
DG-I °	DG	DG-3 & AUX. EQPT.	441' & 455'	3	
DG-I	DG	MC-4A	441' & 455'	3	
DG-I	DG	PP-4A	441' & 455'	3	
DG-I	DG	ELP-4A	441' & 455'	3	
DG-I	DG	TR-4A	441' & 455'	3	
DG-I	DG	DGHV-III	441' & 455'	3	
DG-I	DG	SM-4	441' & 455'	3	
DG-I	DG	TR-4-41	441' & 455'	3	
DG-I	DG	B1-HPCS	441' & 455'	3	
DG-I	DG	C1-HPCS	441' & 455'	3	
DG-I	DG	H22-P028	441' & 455'	3	
DG-II °	DG	DG-1 & AUX. EQPT.	441' & 455'	1	
DG-II	DG	DG-1-7	441' & 455'	1	
DG-II	DG	PP-7A-A-A	441' & 455'	1	
DG-II	DG	TR-7A-A-A	441' & 455'	1	
DG-II	DG	ELP-7A-A	441' & 455'	1	
DG-II	DG	TR-7A-A	441' & 455'	1	
DG-II	DG	MC-7A-A	441' & 455'	1	
DG-II	DG	DGHV-I	441' & 455'	1	
DG-III °	DG	DG-2 & AUX. EQPT.	441' & 455'	2	*
DG-III	DG	DG-2-8	441' & 455'	2	*
DG-III	DG	MC-8A-A	441' & 455'	2	*
DG-III	DG	ELP-8A-A	441' & 455'	2	
DG-III	DG	TR-8A-A	441' & 455'	2	
DG-III	DG	PP-8A-A-A	441' & 455'	2	
DG-III	DG	TR-8A-A-A	441' & 455'	2	
DG-III	DG	DGHV-II	441' & 455'	2	

° See Sh. 2 for additional equipment in these areas.

* Indicates equipment is a part of the dedicated shutdown system.

TABLE 040.75-7

LOCATION OF MAJOR SAFETY RELATED

ELECTRICAL DISTRIBUTION EQUIPMENT

AREA	BLDG	EQUIPMENT	ELEV	DIV
DG-IV	DG	NONE	441'	-
DG-V	DG	NONE	441'	-
DG-VI	DG	NONE	441'	-
DG-VII	DG	NONE	441'	-
DG-VIII	DG	NONE	441'	-
DG-IX	DG	NONE	441'	-
DG-X	DG	NONE	455'	-
DG-II	DG	DP-S1-1E	441'	1
DG-III	DG	DP-S1-2E	441'	2
DG-I	DG	"PNL HPCS"	441' & 455'	3
DG-I	DG	"ENG&GEN AUX LOAD AC PNL"	441' & 455'	3

TABLE 040.75-7

LOCATION OF MAJOR SAFETY RELATED

ELECTRICAL DISTRIBUTION EQUIPMENT

AREA	BLDG	EQUIPMENT	ELEV	DIV	
SW-I	SWIA	IR-26	431' & 441'	2	
SW-I	SWIA	PP-7A-G	431' & 441'	1	
SW-I	SWIA	TR-7A-F	431' & 441'	1	
SW-I	SWIA	PP-7A-B	431' & 441'	1	
SW-I	SWIA	IR-21 & 24	431' & 441'	1 & 3	
SW-I	SWIA	SUPV. PNLS S-1 & S-3	431' & 441'	1 & 3	
SW-II	SWIB	IR-25	431' & 441'	1	
SW-II	SWIB	IR-22	431' & 441'	2	*
SW-II	SWIB	PP-8A-G	431' & 441'	2	*
SW-II	SWIB	PP-8A-B	431' & 441'	2	*
SW-II	SWIB	TR-8A-F	431' & 441'	2	*
SW-II	SWIB	SUPV. PNL S-2	431' & 441'	2	*

TABLE 040.75-7

LOCATION OF MAJOR SAFETY RELATED

ELECTRICAL DISTRIBUTION EQUIPMENT

AREA	BLDG	EQUIPMENT	ELEV	DIV
TG-I	TG	IR-81	471'	
TG-I	TG	IR-82	471'	RPS DIVISIONS
TG-I	TG	IR-83	471'	4 thru 7
TG-I	TG	IR-84	471'	
TG-II	TG	NONE		-
TG-III	TG	NONE		-
TG-IV	TG	NONE		-
TG-V	TG	NONE		-
TG-VI	TG	NONE		-
TG-VII	TG	NONE		-
TG-VIII	TG	NONE		-
TG-IX	TG	NONE		-
TG-X	TG	NONE		-

TABLE 040.75-7

LOCATION OF MAJOR SAFETY RELATED

ELECTRICAL DISTRIBUTION EQUIPMENT

AREA	BLDG	EQUIPMENT	ELEV	DIV
S-I	SERVICE	NONE	-	
S-II	SERVICE	NONE	-	
S-III	SERVICE	NONE	-	
S-IV	SERVICE	NONE	-	
S-V	SERVICE	NONE	-	
S-VI	SERVICE	NONE	-	
S-VII	SERVICE	NONE	-	
CW-I	CIRC. WTR.	NONE	-	
CW-II	PUMPHOUSE	NONE	-	
CT-I	C.T. BLDG #1	NONE	-	
CT-II	C.T. BLDG #2	NONE	-	
-	MAKEUP WTR PUMPHOUSE	NONE	-	
-	COND STOR TNK & YARD AREA	NONE	-	
-	WELL WATER PUMPHOUSES NO. 1, 2, & 3	NONE	-	

TABLE 040.75-7

LOCATION OF MAJOR SAFETY RELATED

ELECTRICAL DISTRIBUTION EQUIPMENT

AREA	BLDG	EQUIPMENT	ELEV	DIV	
RC-I	RW	NONE	437' thru 501'	-	
RC-II	RW	NONE	484'	-	
RC-III	RW	NONE	467' thru 525'	-	
RC-IV	RW	DP-S1-1	467'	1	*
RC-IV	RW	DP-S2-1	467'	1	*
RC-IV	RW	MC-S1-1D	467'	1	*
RC-IV	RW	C2-1	467'	1	
RC-IV	RW	C1-1	467'	1	
RC-IV	RW	C0-1A	467'	1	
RC-IV	RW	C0-1B	467'	1	
RC-IV	RW	PP-7A	467'	1	
RC-IV	RW	TR-7A	467'	1	
RC-IV	RW	MC-7A	467'	1	
RC-IV	RW	IN-3	467'	1	
RC-V	RW	B2-1	467'	1	*
RC-V	RW	B1-6	467'	1	*
RC-V	RW	B0-1A	467'	1	
RC-V	RW	B0-1B	467'	1	
RC-VI	RW	B1-2	467'	2	*
RC-VI	RW	B0-2A	467'	2	
RC-VI	RW	B0-2B	467'	2	
RC-VII	RW	DP-S1-2	467'	2	*
RC-VII	RW	MC-S1-2D	467'	2	*
RC-VII	RW	C1-2	467'	2	
RC-VII	RW	C0-2A	467'	2	
RC-VII	RW	C0-2B	467'	2	
RC-VII	RW	MC-8A	467'	2	*
RC-VII	RW	IN-2	467'	2	
RC-VII	RW	TR-8A	467'	2	
RC-VII	RW	PP-8A	467	2	

TABLE 040.75-7

LOCATION OF MAJOR SAFETY RELATED

ELECTRICAL DISTRIBUTION EQUIPMENT

AREA	BLDG	EQUIPMENT	ELEV	DIV	
RC-VIII	RW	SM-8	467'	2	*
RC-VIII	RW	TR-8-81 & NR-T81	467'	2	*
RC-VIII	RW	SL-81	467'	2	*
RC-VIII	RW	SL-83	467'	2	*
RC-VIII	RW	TR-8-83 & NR-T-83	467'	2	*
RC-IX	RW	C61-P001	467'	1 & 2	*
RC-IX	RW	BD - "RS"	467'	1 & 2	*
RC-IX	RW	PP-7A-F	467'	1	*
RC-IX	RW	DP-S1-1D	467'	1	*
RC-IX	RW	DP-S1-2D	467'	2	*
RC-IX	RW	PP-8A-F	467'	2	*
RC-X	RW	-See Note #1-	501'	See Note #1	
RC-XI	RW	COHV-1	525'	1	
RC-XI	RW	COHV-3	525'		
RC-XI	RW	WOA-SR-18A	525'		
RC-XI	RW	WOA-SR-18B	525'		
RC-XI	RW	MC-7F	525'	1	
RC-XII	RW	MC-8F	525'	2	*
RC-XII	RW	COHV-2	525'	2	*
RC-XII	RW	COHV-4	525'		
RC-XII	RW	WOA-SR-19A	525'		
RC-XII	RW	WOA-SR-19B	525'		
RC-XIII	RW	NONE	484', 501', 525'		
RC-XIV	RW	SM-7	441'	1	
RC-XIV	RW	TR-7-71 & NR-T71	467'	1	
RC-XIV	RW	SL-73	467'	1	
RC-XIV	RW	SL-71	467'	1	
RC-XIV	RW	TR-7-73 & NR-T73	467'	1	

TABLE 040.75-7

LOCATION OF MAJOR SAFETY RELATED

ELECTRICAL DISTRIBUTION EQUIPMENT

AREA	BLDG	EQUIPMENT	ELEV	DIV
RC-XV	RW	NONE	437' thru 501'	
RC-XVI	RW	NONE	437' thru 525'	
RC-XVII	RW	NONE	437' thru 501'	
RC-XVIII	RW	NONE	467' & 484'	
RC-XIX	RW	NONE	467'	

TABLE 040.75-7

LOCATION OF MAJOR SAFETY RELATED

ELECTRICAL DISTRIBUTION EQUIPMENT

AREA	BLDG	EQUIPMENT	ELEV	DIV	
R-I	RB	MC-7B	522'	1	
R-I	RB	ELP-7B-B	471'	1	
R-I	RB	ELP-7B-A	606'	1	
R-I	RB	PP-7A-E	471'	1	
R-I	RB	PP-8A-E	471'	2	
R-I	RB	SH-9	471'	1	
R-I	RB	SH-10	471'	2	
R-I	RB	SH-11	522'	2	
R-I	RB	SH-12	522'	1	
R-I	RB	MC-7B-A	522'	1	*
R-I	RB	MC-7B-B	572'	1	
R-I	RB	ELP-8B-A	606'	2	
R-I	RB	IR-61	422'	1	
R-I	RB	IR-62	471'	1	
R-I	RB	IR-63	501'	2	
R-I	RB	IR-64	501'	2	
R-I	RB	IR-65	471'	1	
R-I	RB	IR-66	501'	1	
R-I	RB	IR-67	548'	1	
R-I	RB	IR-68	548'	2	*
R-I	RB	IR-69	522'	2	
R-I	RB	IR-70	522'	2	
R-I	RB	IR-71	522'	1	
R-I	RB	IR-73	522'	1	
R-I	RB	IR-74	522'	2	
R-I	RB	H22-P001	471'		
R-I	RB	H22-P002	522'		
R-I	RB	H22-P004	522'	1	
R-I	RB	H22-P005	522'		

TABLE 040.75-7

LOCATION OF MAJOR SAFETY RELATED ELECTRICAL DISTRIBUTION EQUIPMENT

AREA	BLDG	EQUIPMENT	ELEV	DIV	
R-I	RB	H22-P006	471'		
R-I	RB	H22-P007	522'		
R-I	RB	H22-P009	471'		
R-I	RB	H22-P010	471'		
R-I	RB	H22-P011	548'		
R-I	RB	H22-P015	501'		
R-I	RB	H22-P017	471'		*
R-I	RB	H22-P018	501'	1	
R-I	RB	H22-P021	501'	2	
R-I	RB	H22-P022	471'		*
R-I	RB	H22-P024	471'		
R-I	RB	H22-P025	501'		
R-I	RB	H22-P026	522'		*
R-I	RB	H22-P027	522'		
R-I	RB	H22-P029	471'		
R-I	RB	H22-P030	501'		
R-I	RB	H22-P031	501'		
R-I	RB	H22-P032	501'		
R-I	RB	H22-P033	501'		
R-I	RB	CMS-SR-13	548'	1	
R-I	RB	CMS-SR-14	548'	2	
R-I	RB	CMS-SR-20	548'		
R-I	RB	CMS-SR-21	548'		
R-I	RB	CMS-SR-32A	548'		
R-I	RB	CMS-SR-32B	548'		
R-I	RB	CMS-SR-33A	548'		
R-I	RB	CMS-SR-33B	548'		
R-I	RB	CRD-IR-1A	422'		
R-I	RB	CRD-IR-1B	422'		

TABLE 040.75-7

LOCATION OF MAJOR SAFETY RELATED

ELECTRICAL DISTRIBUTION EQUIPMENT

AREA	BLDG	EQUIPMENT	ELEV	DIV
R-I	RB	CRD-IR-1C	422'	
R-I	RB	CRD-IR-1	522'	
R-I	RB	CRD-IR-2	522'	
R-I	RB	CRD-IR-3	522'	
R-I	RB	ELP-8B-B	471'	2
R-I	RB	TR-7B-A	606'	1
R-I	RB	TR-8B-A	606'	2
R-I	RB	TR-8B-B	471'	2
R-I	RB	TR-7B-B	471'	1
R-I	RB	SGT-EHC-1A1 CP	572'	1
R-I	RB	SGT-EHC-1B1 CP	572'	1
R-I	RB	SGT-FU-1A CP	572'	1
R-I	RB	CAC-HR-1A CP	572'	1
R-II thru R-XVII	RB	NONE		
R-XVIII	RB	MC-8B	522'	2 *
R-XVIII	RB	MC-8B-A	522'	2 *
R-XIX	RB	MC-8B-B	572'	2 *
R-XIX	RB	SGT-EHC-1A2 CP		2
R-XIX	RB	SGT-EHC-1B2 CP		2
R-XIX	RB	CAC-HR-1B CP		2
R-XIX	RB	SGT HUMID TRANS PNL		2
R-XIX	RB	SGT-FU-1B		
R-XX	RB	MC-S2-1A	471'	1 *
R-XX	RB	VB-1A	471'	1

TABLE 040.75-7

NOTES:

#1 Safety Related Equipment in the Main Control Room is too numerous to provide a detailed listing. They include the following:

NSSS Control Boards * (Div. 1 thru 7)
BOP Control Boards * (Div. 1 thru 3)
Relay Panel RC-1 and RC-2 * (Div. 1 & 2)
Supervisory Cabinets CS-1, 2*, 3 (Div. 1 thru 3)
DP-SO-1A & DP-SO-1B (Div. 1 & 2)
PP-7A-A & PP-8A-A * (Div. 1 & 2)
DP-S1-1A * & DP-S1-2A * (Div. 1 & 2)
Instrument Racks COHV-5A & 5B *

#2 This equipment connects to Mechanical, Instrumentation, and HVAC equipment that is part of the dedicated shutdown system

Equipment is Either:

Power Ckts:

a. Directly connected to the components

b. Part of the power train from the equipment to the dc source (Div. 2) or battery source (Div. 1)

Control Ckts:

Part of the complete control circuit required for device operation.

TABLE 040.75-8

EQUIPMENT	CABLE NO's	CA TYPE	POWER SOURCE	ELECTRICAL DRAWINGS ONE-LINE	WIRING	BLOCK	REMARKS
RCIC-P-1							Turbine Driven Pump
(E51-C001)							
Controls							
RCIC-P-2	1M21A-180	P	MC-S2-1A	E505	2-22-2205	E878-49	
(E51-C002)	181	C	DP-S2-1		E519-34A		
	182	C	BATT B2-1				
RCIC-P-4	1M21A-190	P	MC-S2-1A	E505	2-22-2205	E878-49	
(E51-C004)	191	C	DP-S2-1		E519-34A		
	192	C	BATT B2-1				
RCIC-V-1	1M21A-10	P	MC-S2-1A	E505	2-22-2206	E880-2	Turb Trip/Throttle MOV
(w/E51-C002)	11,12,13	C	DP-S2-1			F878-49	
	14,15,16	C	BATT B2-1				
			BATT B1-1				
	1RC1C-27	C	MC-S1-1D		2-22-2203		(1D11A-9 DP-S1-1A)
			DP-S1-1				
			HB-P621 Bus A				
RCIC-V-2	1RC1C-31		MC-S1-1D		2-22-1209		Hydraulic Oper Vlve
(E51-C002)	32		DP-S1-1		2-22-2206		(Turb gov valve) Mtd
			:		2-42-0203		on Trip & Throttle V
			BATT B1-1		E540-8		
RCIC-V-8	1M11D-50	P	MC-S1-1D	E505	2-22-2206	E878-51	
(E51-F008)	51	C	DP-S1-1		E519-34A	E880-2	
	52	C					
	53	C	BATT B1-1		F522		
	54	C					
RCIC-V-10	1M11D-30	P	MC-S1-1D	E505	2-22-2206	E878-51	
(E51-F010)	31	C	DP-S1-1		F519-34A	E880-2	
	32	C	BATT B1-1				
	33	C					

TABLE 040.75-8

EQUIPMENT	CABLE NO's	CA TYPE	POWER SOURCE	ELECTRICAL DRAWINGS ONE-LINE	WIRING	BLOCK	REMARKS
RCIC-V13	1M21A-120	P	MC-S2-1A	E505	2-22-2206	E878-49	
(E51-F013)	121	C	DP-S2-1		E522	F880-2	
	122	C					
	124	C	BATT B2-1				
	125	C					
RCIC-V-19	1M21A-130	P	MC-S2-1A	E505	2-22-2206	E878-49	
(E51-F019)	131	C	DP-S2-1		F519-34A	E880-2	
	132	C					
	134	C	BATT B2-1		F522		
	135	C					
RCIC-V-22	1M21A-140	P	MC-S2-1A	E505	2-22-2206	F878-49	
(E51-F022)	141	C	DP-S2-1			E880-2	
	142	C					
	143	C	BATT B2-1				
RCIC-V-31	1M11D-10	P	MC-S1-1D	E505	2-22-2206	F878-51	
(E51-F031)	11	C	DP-S1-1		F519-34A	E880-2	
	12	C					
	13	C	BATT B1-1		F522		
	IRC1C-28	C			E537-28		1D11A-9
RCIC-V-45	1M21A-170	P	MC-S2-1A	E505	2-22-2206	F878-49	
(E51-F045)	171	C	DP-S2-1				
	172	C	BATT B2-1		E519-24	E880-2	
	173	C			E537-7A		
	174	C	MC-S1-1D		E519-34A		
			DP-S1-1				
	1RC1C-24,34	C	BATT B1-1	E540-6	2-22-2202		1D11A-9
RCIC-V-46	1M11D-170	P	MC-S1-1D	E505	2-22-2206	E878-51	See PED 218-E-2137
(E51-F046)	160	C	DP-S1-1		F519-34A	E880-2	
	1	C	BATT B1-1				
RCIC-V-63	2MBBA-360	P	MC-88-A	E503-7	2-22-2206	E878-36	
(E51-F063)	361	C	MC-8B		E519-34A	E880-2	
	362		SL-81		E522		
	363		DG-2.		2-42-0204		
	364						
	365						
	366						

TABLE 040.75-8

EQUIPMENT	CABLE NO's	CA TYPF	POWFR SOURCF	ELECTRICAL DRAWINGS ONE-LINE	WIRING	BLOCK	REMARKS
RCIC-V-64 (E51-F064)	1M21A-160	P	MC-S2-1A	E505	2-22-2206	F878-49	
	161	C	DP-S2-1		E519-34A	E880-2	
	162	C	BATT B2-1		E522		
	163	C					
	164	C					
RCIC-V-68 (E51-F068)	1M11D-150	P	MC-S1-1D	E505	2-22-2206	E878-49	
	151	C	DP-S1-1		E519-34A	E880-2	
	152	C	BATT B1-1		E522		
	153	C					
RCIC-V-69 (E51-F069)	1M21A-200	P	MC-S2-1A	E505	2-22-2206	E878-49	
	201	C	DP-S2-1		E519-34A	E880-2	
	202	C	BATT B2-1		F522		
	203	C					
	204	C					
	207	C					
	208	C					
RCIC-V076 (E12-F076)	2M8BA-440,441	P	MC-8B-A	E503-7	2-22-2206	E878-36	
	442	C	MC-8B		F519-1	F880-2	
	443	C	SL-81		E522		
	444	C	DG-2		E539-40		
	BRC1C-9010	C					Alarm
RCIC-HX-1	--	--	--	--	--	--	Heat Exchanger No Direct Elect. Connections
RCIC-TK-1	--	--	--	--	--	E880-2	Tank No Direct Elect. Connections
RCIC-VT-1	--	--	--	--	--	--	Steam Turbine No Direct Elect. Connections

TABLE 040.75-8

EQUIPMENT	CABLE NO's	CA TYPE	POWER SOURCE	ELECTRICAL DRAWINGS ONE-LINE	WIRING	BLOCK	REMARKS
RCIC	IRC1C-29	C		E505	F539-22	E880-2	Hi Lev. Cont
Barometric					E537-7A		
Condenser Pkg			MC-S1-1D				
			DP-S1-1				
			BATT B1-1		2-22-2203		H13-P621 Bus A
RCIC-FE-1	--	--	--	--	--	--	Orifice
(E51-N001)							No Electrical Connections
RCIC-FT-3	IRC1C-35	S	RS BD		2-22-2205		H22-P017-H13-P612
(E51-N003)	36	S	MC-S1-1D	E505	2-42-0203		Indic.
			DP-S1-1		E539-16		
			BATT B1-1		E539-28		AD11A-9005 (RC1C-35)
RHR-P-2B	2SM8-50	P	SM-8	E502-2	E517-13	E876-5	
(E12-C002)	51	P	DG-2		2-17-0705	E882-4	
					2-17-0715		
	2SM8-54,55	C					
	56,57						
RHR-V-3B	2M8BB-140	P	MC-8B-B	E503-12	E519-34A	E878-53	
(E12-F003B)	141	C	MC-8B		2-17-0713	E880-10	
	142	C	SL-81				
	143		DG-2				
RHR-V-4B	2M8BA-20	P	MC-8B-A	E503-7	E522	E878-34	
(E12-F004B)	22	C	MC-8B		2-17-0713	E878-36	
	23	C	SL-81		2-42-0207	E880-10	
	24	C	DG-2				
	17	C					

TABLE 040.75-8

EQUIPMENT	CABLE NO's	CA TYPE	POWER SOURCE	ELECTRICAL DRAWINGS ONE-LINE	WIRING	BLOCK	REMARKS
RHR-V-6B (E12-F006B)	2M8BA-10	P	MC-8B-A	E503-7	E519-1	E878-36	
	12,13	C	MC-8B				
	14,15,16	C	SL-81, DG-2		E519-34A	E880-10	
	17,18	C	DP-S1-2D	E50	2-17-0713		
	2RHR-41	C	DP-S1-2				2DI2D-5 C61 Pool Supply
	BRHR-9035,9036	C	BATT B1-2		2-42-0207		Alarms
RHR-V-11B (E12-F011B)	2M8BA-230	P	MC-8B-A	E503-7	E519-34A	E878-36	
	232	C	MC-8B		2-17-0713	E880-10	
	233	C	SL-81		2-42-0207		
	234	C	DG-2				
RHR-V16B (E12-F016B)	2M8BA-140	P	MC-8B-A	E503-7	E519-34A	E878-36	
	142	C	MC-8B		E522	E880-10	
	143	C					
	144	C	SL-81		2-17-0713		
	132	C	DG-2		2-17-0715		
RHR-V-23 (E12-F023)	1M21A-40	P	MC-S2-1A	E505	E522	E878-49	
	41	C	DP-S2-1		2-06-0907	E880-2	
	42	C	BATT B2-1				
	43	C					
RHR-V-24B (E12-F024B)	2M8BA-90	P	MC-8B-A	E503-7	E519-34A	E878-36	
	92	C					
	93	C	MC-8B		E522	E880-10	
	94	C	SL-81		2-17-0713		
	16	C	DG-2		2-42-0208		
RHR-V-26B (E12-F026B)	2M8BA-160	P	MC-8B-A	E503-7	E519-34A	E878-36	
	162	C	MC-8B		2-17-0713	E880-10	
	163	C	SL-81		2-42-0208		
	164	C	DG-2				
RHR-V-27B	2M8BA-150	P	MC-8B-A	E503-7	E519-34A	E878-36	
	152	C	MC-8B		E522	E880-10	
	153	C	SL-81		2-17-0706		
	154	C	DG-2		2-17-0713		
	15	C					

TABLE 040.75-8

EQUIPMENT	CABLE NO's	CA TYPE	POWER SOURCE	ELECTRICAL DRAWINGS ONE-LINE	WIRING	BLOCK	REMARKS
RHR-V-42B	2M8BA-100	P	MC-8B-A	E503-7	E519-34A	E878-36	
(E12-F042B)	102	C	MC-8B	E508-1	F522	F880-10	
	103	C			2-17-0705		
	104	C	PP-8A-A		2-17-0714		
	2RHR-78		DG-2	E537-6A	2-42-0208		2P8AA-8
RHR-V-47B	2M8BB-120	P	MC-8B-B	E503-12	E519-34A	E878-53	
(E12-F047B)			MC-8B		2-17-0713	E880-10	
	2M8BB-121	C	SL-81		2-42-0209		
	122,123		DG-2				
RHR-V-48B	2M8BB-130	P	MC-8B-B	E503-12	E519-1	E878-53	
(E12-F048B)			MC-8B		E519-34A	E880-10	
	2M8BB-131	C	SL-81		2-17-0713		
	132,133		DG-2		2-42-0209		
RHR-V-49	2M8BB-230	P	MC-8B-B	E503-12	E519-1	F878-53	
(E12-F-49)			MC-8B		2-06-0907	F880-2	
	2M8BB-231	C	SL-81		2-42-0209		
	232,233,234		DG-2				
RHR-V-52B	2M8BB-250	P	MC-8B-B	E503-12	F519-1	F878-53	
(E12-F052B)			MC-8B		F519-34A	E880-10	
	2M8BB-251	C	SL-81		2-17-0713		
	252,253		DG-2		2-42-0208		
RHR-V-53B	1M7BA-580	P	MC-7B-A	E503-7	F519-1	F878-28	
(E12-F053B)	581	C	MC-7B		E519-34A	E880-10	Per PED
	582	C	SL-71				218-F-1759
	583	C	SM-7		2-42-0208		
			DG-7		2-07-0714		
RHR-V-60B	2NS4-1,29	C		E504	E537-6A	E880-2	
(E12-F060B)					E537-6C		
					2-06-0908		
					2-06-0907		
					E539-21		Power Supply

TABLE 040.75-8

EQUIPMENT	CABLE NO's	CA TYPE	POWER SOURCE	ELECTRICAL DRAWINGS ONE-LINE	WIRING	BLOCK	REMARKS
RHR-FCV-64B	2M8BA-390	P	MC-8B-A	E503-7	E522	E878-36	
(E12-F064B)	391	C	MC-8B		2-17-0714	E880-10	
	392	C					
	BRHR-9036		56-81		E540-11		Alarm
			DG-2				
RHR-V-68B	2M8BB-220	P	MC-8B-B	E503-12	E519-34A	E878-53	
(E12-F068B)			MC-8B		2-17-0714	E880-10	
	2M8BB-221	C	SL-81		2-42-0210	E876-5	
	222,223,224		DG-2				
RHR-V-74B	2M8BB-150	P	MC-8B-8	E503-12	2-17-0714	E878-53	
(E12-F074B)	151	C	MC-8B			E880-10	
	152	C	SL-81				
	2RHR-34	C	DG-2				Pos Indicator
RHR-V-115	2M8BB-270	P	MC-8B-B	E503-12	2-17-0713	E878-53	
(E12-F094)			MC-8B			E880-10	
	2M8BB-271	C	SL-81				
	272	C	DG-2				
RHR-HX-1B	--	--	--	--	--	--	Heat Exchanger
							No Direct Elect. Connections
RHR-FE-14B			H13-P613				Non Elect Per Flow
(E12-N014B)	2RHR-35	S	PP-8A-A	E504			(Actuates RHR-FT-15B
RHR-FT-15B			MC-8A		2-17-0707	M521-H5	
			SL-83				
			SM-8				
			DG-2				2P8AA-006

TABLE 040.75-8

EQUIPMENT	CABLE NO's	CA TYPE	POWER SOURCE	ELECTRICAL DRAWINGS ONE-LINE	WIRING	BLOCK	REMARKS
SW-P-1B	2SM8-80	P	SM-8 DG-2	E502-2	E517-14	F876-5 E886-5	
	2SM8-81,84 85,86,87, 88,89,93	C					
	BSM8-9089	C					Mislabelled CA should be Div 2
SW-V-2B	2M8A-40	P	MC-8A	E503-6	E527-7	F878-32	
	2M8A-41,42 43,44,45, 46,47	C	SL-83 DG-2		E527-6	E886-5	
SW-V-4B	2M8AA-160	P	MC-8A-A MC-8A	E503-9	E519-9	E878-33 E883-2	
	2M8AA-161 162,163	C	SL-83 DG-2				
SW-V-6B-1 (SW-V-216)	2DG2-19	C	See Remarks	53-00-0054	--	F883-2	Furnished as part of Vendor Pkg. Powered from same source as other DG components.
SW-V-6B-2 (SW-V-217)	2DG2-21	C	See Remarks	53-00-0055	--	F883-2	Furnished as part of Vendor Pkg. Powered from same source as other DG components.
SW-TCV-11B	2COV2-20,21	C	COHU-2,5 PP-8A-F	E508-1	E519-18	F885-4	Per PED 218-E-1782, SW-TCV-11B contains
SW-PS-11B	2COV2-22,26	C	PP-8A-A MC-8A		E538-28 E537-22A	E885-5 E878-41	Int'les from SW-PS-11B & WMA-TIC-11B
WMA-TIC-11B	2COV5-3	C	SL-83 SM-8 DG-2		E535-61C		

TABLE 040.75-8

EQUIPMENT	CABLE NO's	CA TYPE	POWER SOURCE	ELECTRICAL DRAWINGS ONE-LINE	WIRING	BLOCK	REMARKS
SW-V-12B	2M8A-50	P	MC-8A SL-83	E503-6	E527-7 E527-6	F878-32 E886-5	
	2M8A-51,52,53 54,55,56	C	DG-2				
SW-V-24B	2M8BA-280	P	MC-8B-A MC-8B	E503-7	F519-9	E878-36 F878-32	
	2M8BA-281, 282,283,284	C	SL-81 DG-2			E885-17	
SW-V-34	AP7CA-9108	C	PP-7CA-A	E508-2	E519-9 E535-65E	E878-49	Sol. Vlv.
	ARC1C-9051 9052	C					
SW-PCV-38B	BIR22-9060	C	IR-22 PP-8A-A		E527-8	E886-5	IR-22
	9061		PP-8A-G MC-8A		E536-6C	F885-23	
	9062		SL-83 DG-2				
SW-V-69B	2M8A-60 61,62,63,64 65,67,68	P C C	MC-8A SL-83 DG-2	E503-6	E527-7	F878-32 E886-5	
SW-V-75B	2M8BA-490	P	MC-8B-A MC-8B	E503-7	F519-25A	E878-36A	PED 5192 Fuel Pool Upgrading
	2M8BA-491 492	C	SL-81 DG-2				
SW-V-90	2M8AA-320	P	MC-8A-A MC-8A	E503-9	E519-20	E878-33	
	2M8AA-321,322 323	C	SL-83 DG-2				

TABLE 040.75-8

		CA	POWER	ELECTRICAL DRAWINGS			
EQUIPMENT	CABLE NO's	TYPE	SOURCE	ONE-LINE	WIRING	BLOCK	REMARKS
SW-V-187B	2M8BA-570	P	MC-8B-A MC-8B	E503-7	E519-25A	E878-36A	PFD 218-F-5192
	2M8BA-571,572	C	SL-81 DG-2				
SW-V-188B	2M8BB-320	P	MC-8B-B MC-8B	E503-12	E519-25A	E878-53A	PED-218-F-5192
	2M8BB-321 322	C	SL-81 DG-2				
SW-PT-32BR	2IR22-1	S	PP-8A-G PP-8A-B MC-8A SL-83		M634 24-RS53	E885-29	IR-22 & PNL 52
(For supv PNL S2)	2MISC-1	S	SM-8 DG2-8				Data Bus Cable
SW-PI-32BR	BMISC-9200	S	PP-8A-A MC-8A SL-83		M634 24-RS53	F886-5	BD RS, CS2
(For supv PNL CS2)	2MISC-150	S	SM-8 DG-2				Data Bus Cable
SW-TE-1BR (Local)	ACOMA-9254	S	PP-8A-G PP-8AB MC-8A SL-83 SM-8 DG-2		M634- 24-RS52	F886-5	Supv PNL S2 (Analog cables are "A")
SW-TI-1BR	BMISC-9200	S	DP-S1-2D DP-S1-2 BATT B1-2 PP-8A-F PP-8A-A MC-8A SL-83,SM-8,D52-8		M634- 24-RS52	F886-5	BD 'RS'
SW-LTD-1BR	BIR22-9077	S	PP-8A-G PP-8A-B DG-2		M634- 24-RS51	E885-23	IR-22 CA by vendor

TABLE 040.75-8

EQUIPMENT	CABLE NO's	CA TYPE	POWER SOURCE	ELECTRICAL DRAWINGS ONE-LINE	WIRING	BLOCK	REMARKS
SW-LT-1BR	BIR22-9078 BIR22-9052	S C	PP-8A-G PP-8A-B MC-8A SL-83 SM-8 DG2-8		M634- 24-RS51	E886-5	IR-22
SW-LI-1BR	BMISC-9200	S	DP-S1-2D DP-S1-2 BATT B1-2 PP-8A-F PP-8A-A MC-8A,51-83 SM-B,DG-2		M634- 24-RS51	F886-5	BD 'RS'
MS-RV-4A (B22-F013S)	2ADS-29	C2	H13-P628 Bus "B" DP-51-2A DP-51-2 BATT B1-2	E509	2-06-1006 E539-4,38	F880-1	2D12A-5
MS-RV-4C (B22-F013M)	2ADS-25	C	↓	E509	↓	↓	
MS-RV-4D (B22-F013P)	2ADS-27	C	↓	E509	↓	↓	
MS-V-22A (B22-F022A)	2NS4-11,21,22	C		E504	2-06-0909 2-06-0904 E539-37,36 21	F880-2	
MS-V-22B (B22-F022B)	2NS4-9,19,20	C		↓	↓	↓	

TABLE 040.75-8

EQUIPMENT	CABLE NO's	CA TYPE	POWER SOURCE	ELECTRICAL DRAWINGS ONE-LINE	WIRING	BLOCK	REMARKS
MS-V-22C (B22-F022C)	2NS4-10,17,18	C		E504	2-06-0909 2-06-0904 E539-37,36	E880-2	
MS-V-22D (B22-F022D)	2NS4-8,15,16	C					
MS-LIS-37A (B22-LIS-N037A)	IRCIC-18	C C	DP-S1-1A DP-S1-1D MC-S1-1D BATT B1-1	E504	2-22-2202 2-06-1004		H22-P026 ID11A-8
MS-LIS-37B (B22-LIS-N037B)	2RHR-11 2ADS-40 2RCIC-14	C C C	H13-P618 Bus "B" PP-8AA DP-S1-2A		2-17-0705 2-06-1004 E539-3 2-22-2204	F880-10 F880-1	BP8AA-9008

TABLE 040.75-8

EQUIPMENT	CABLE NO's	CA TYPE	POWER SOURCE	ELECTRICAL DRAWINGS ONE-LINE	WIRING	BLOCK	REMARKS
C61-FIC-R001	(Internal)		RS BD DP-S1-1D	E509	2-42-0203		(C61-P001) RS BD 2-42-0201 1D11D
C61-SQRT-K001	(Internal) Init Cables 1RC1C-35 36	 S S	RS BD DP-S1-1D		2-42-0203		1D11D
C61-FI-R001-1			RS BD DP-S1-1D				1D11D
C61-SI-R003	(Internal) Init. Cable 1RC1C-15,41 1RC1C-14		RS BD DP-S1-1D				1D11D
C61-LITS-N026D (MS-LITS-26D) (B22-N026D)	6NS4-2 ANS4-9003	S S	RS BD PP-7A-F	E508-1	2-06 -0925		1P7AF
C61-LI-R010	Internal		RS BD PP-7A-F				1P7AF
C61-PT-N006	ANS4-9004	S	RS BD PP-7A-F				

040.075-41

WNP-2

TABLE 040.75-8

EQUIPMENT	CABLE NO's	CA TYPE	POWER SOURCE	ELECTRICAL DRAWINGS ONE-LINE	WIRING	BLOCK	REMARKS
C61-PI-R011	Internal		RS BD PP-7A-F	E508-1	2-42-0203		(C61-P001) RS BD 1P7AF-1
C61-FT-N001	1RHR-60	S	RS BD PP-7A-F	 E508-1	2-42-0203		(C61-P001) RS BD
C61-FT-R005	Internal		RS BD PP-7A-F	E508-1	2-42-0203		(C61-P001) RS BD 2P8AF-1 1P7AF-1
CMS-PT-2R	BMISC-9201	S	RS BD PP-7A-F	E508-1	M634-43- RS53	E885-32	IR-68
CMS-LT-2R (Local)	BMISC-9202	S	RS BD PP-7A-F	E508-1	M634-43 RS60 E538-40	E885-17	BD 'RS"
CMS-PI-2R	Initiating Cable BMISC-9201		RS BD PP-7A-F	E508-1	M634-43- RS58	E885-32	BD 'RS'

TABLE 040.75-8

EQUIPMENT	CABLE NO's	CA TYPE	POWER SOURCE	ELECTRICAL DRAWINGS ONE-LINE	WIRING	BLOCK	REMARKS
CMS-LI-2R	Initiating Cable BMISC-9202	S	BD RS PP-8A-F PP-8A-A MC-8A SL-83 SM-8 DG-2	E508-1	M634-43 RS60	F885-17	BD 'RS' 2P8AF-2
CMS-PT-6R	BMISC-9201	S	BD RS PP-8A-F	E508-1	M634-43- RS59	F885-17	IR-68
CMS-PI-6R	↓	S	BD RS PP-8A-F	E508-1	M634-43- RS59	F885-17	BD 'RS'
CMS-TE-19R	2CACS-331 322 BCACS-9250		BD RS PP-8A-F	F508-1	M634-43- J53 & 43-RS-51	E888-6 F885-17	TCG2 Via TB.R315
CMS-MV/I-19R	Initiating BCACS-9250	S	BD RS PP-8A-F	E508-1	M634-43- J54 & 43-RS-51		BD 'RS'
CMS-TI-19R	Initiating BCACS-9250	S	BD RS PP-8A-F	E508-1	M634-43- RS51	E885-17	BD 'RS'
CMS-TE-37R	1CACS-332 ACACS-9201		BD RS PP-8A-F	E508-1	M634-43- RS52	E888-6 E885-17	Via-TB-R314

TABLE 040.75-8

EQUIPMENT	CABLE NO's	CA TYPE	POWER SOURCE	ELECTRICAL DRAWINGS ONE-LINE	WIRING	BLOCK	REMARKS
CMS-MV/I-37R	ACACS-9201	S	BD RS PP-8A-F		M634-43 RS52		BD 'RS'
CMS-TI-37R (Internal)	Init. ACACS-9201		BD RS PP-8A-F		M634-43- RS52		BD 'RS'
CMS-TE-39R	2CACS-9250 BCACS-9250	S S	BDG2 BDRS		M634-43 RS53	F888-6 E885-17	BD RS
CMS-MV/I-39R	Initiating BCACS-9250	S	BD RS PP-8A-F		M634-43- RS53		BD 'G2'
CMS-TI-39R	Initiating BCACS-9250		BD RS PP-8A-F		M634-43- RS53		BD 'RS'
CMS-TE-41R	ACACS-9203	S	BD RS PP-8A-F		M634-43- RS-54	F888-6 F885-17	BD RS
CMS-MV/I-141R	Initiating ACACS-9203		BD RS PP-8A-F		M634-43 RS54		BD 'RS'

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TABLE 040.75-8

EQUIPMENT	CABLE NO's	CA TYPE	POWER SOURCE	ELECTRICAL DRAWINGS ONE-LINE	WIRING	BLOCK	REMARKS
CMS-TI-41R	Initiating ACACS-9203	 S	BD RS PP-8A-F		M634-43- RS54		BD 'RS'
CMS-TE-42R	2CACS-143 BCACS-9251	S S	BD RS PP-8A-F		M634-43- J64 E545-13	F888-6 E885-17	 BD 'RS'
CMS-MV/I-42R	Initiating BCACS-9251		BD RS PP-8A-F		M634-43- RS57		BD 'RS'
CMS-TI-42R	Initiating BCACS-9251		BD RS PP-8A-F		M634-43- RS57		BD 'RS'
CMS-TE-43R	2CACS-144 BCACS-9250	S	BD RS PP-8A-F		M634-43- RS-55 E545-13	E888-6 E885-17	
CMS-MV/I-43R	Initiating BCACS-9250	 S	BD RS PP-8A-F		M634-43 RS55		BD 'RS'
CMS-TI-43R	Initiating BCACS-9250	 S	BD RS PP-8A-F		M634-43- RS55		BD 'RS'

TABLE 040.75-8

EQUIPMENT	CABLE NO's	CA TYPE	POWER SOURCE	ELECTRICAL DRAWINGS ONE-LINE	WIRING	BLOCK	REMARKS
CMS-TE-44R	ACACS-9203	S	RS BD PP-8A-F		M634-43- RS56	E882-6 E885-17	
CMS-MV/I-44R	Initiating ACACS-9203		RS BD PP-8A-F		M634-43- RS56		BD 'RS'
CMS-TI-44R	Initiating ACACS-9203		RS BD PP-8A-F		M634-43- RS56		BD 'RS'
RRA-FN-3	2M8B-150	P	MC-8B SL-81	E503-8	E519-12 E521-20	E878-35 E876-5	
	2M8B-151,152	C	SM-8 DG-2				
RRA-FN-6	2M8B-450	P	MC-8B SL-81	E503-8	E519-12 E521-20	E878-35	
	2M8B-451	C	SM-8				
	452	C	DG-2				
RRA-FN-10	2M8B-100	P	MC-8B SL-81	E503-8	E519-13	E878-35	
	2M8B-101	C	SM-8				Includes DMPR Cont. ROA-AD-10
	102	C	DG-2				
	103	C					
WMA-FN-51B	2M8F-20	P	MC-8F SL-83	E503-11	E519-17	E878-41	
	2M8F-21	C	SM-8 DG-2				

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TABLE 040.75-8

EQUIPMENT	CABLE NO's	CA TYPE	POWER SOURCE	ELECTRICAL DRAWINGS ONE-LINE	WIRING	BLOCK	REMARKS
WMA-FN-52B	2M8F-30	P	MC-8F SL-83	E503-11	E519-17	E878-41 F885-5	
	2M8F-31,32	C	SM-8 DG-2				
WMA-AD-52-1	2COV2-50,51	C	CONV-2 PP-8A-F		E519-17 E508-1	E885-5	2P8AF-4
WMA-AD-52-2	2COV2-55,56	C	CONV-2 PP-8A-F		E519-17 E508-1	F885-5	2P8AF-4
WMA-FN-53B	2M8F-50	P	MC-8F SL-83	E503-11	E519-17	E878-41 E885-5	
	2M8F-51,55	C	SM-8 DG-2				
WMA-TIC-11B	2COV5-3 See SW-TCV-11B	C	CONV-5 PP-8A-A		E519-18		2P8AA-110
PRA-FN-1B	2M8AA-180	P	MC-8A-A MC-8A	E503-9	E527-10	F878-33 E876-5 E886-5	
	2M8AA-181,182 183	C	SL-83 SM-8 DG-2				
DMA-FN-22	2M8AA-20	P	MC-8A-A MC-8A	E503-9	E519-20	E878-33 E885-7	
	2M8AAC	C	SL-83 SM-8 DG-2				

TABLE 040.75-8

EQUIPMENT	CABLE NO's	CA TYPE	POWER SOURCE	ELECTRICAL DRAWINGS ONE-LINE	WIRING	BLOCK	REMARKS
DMA-FN-21	2M8AA-10	P	MC-8A-A	E503-9	E519-20	E878-33	
			MC-8A			E885-7	
			SL-83				
	2M8AA-11	C	SM-8				
			DG-2				
DEA-FN-22	2M8AA-60	P	MC-8A-A	E503-9	E519-21	E878-33	
			MC-8A			E885-7	
			SL-83				
	2M8AA-61	C	SM-8				
			DG-2				
DEA-FN-21	2M8AA-30	P	MC-8A-A	E503-9	E519-20	E878-33	
			MC-8A			E885-7	
			SL-83				
	2M8AA-31,32	C	SM-8				
			DG-2				
RRA-FN-14	2M8BB-110	P	MC-8B-B	J03-12	E519-13		
w/ROA-SPV-14	111	C	MC-8B				
	112	C	SL-81				
			SM-8				
	113	C	DG-2				

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TABLE 040.75-8

EQUIPMENT	CABLE NO's	CA TYPE	POWER SOURCE	ELECTRICAL DRAWINGS ONE-LINE	WIRING	BLOCK	REMARKS
E51-N009A	1RC1C-19		H13-P621 Bus A MC-S1-1D DP-S1-1 BATT B1-1		2-22-2202		
E51-N009B	1RC1C-20		H13-P621 Bus A		2-22-2202		
E51-N006	1RC1C-21		H13-P621 Bus A		2-22-2202		
B22-N024D	1RC1C-33		H13-P621 Bus A		2-22-2202		

TABLE 040.75-8

EQUIPMENT	CABLE NO's	CA TYPE	POWER SOURCE	ELECTRICAL DRAWINGS ONE-LINE	WIRING	BLOCK	REMARKS
BATT B1-1							
DP-S1-1	1D11-1, 1D11-11	P		E505			
MC-S1-1D	1D11-3	P		E505			
BATT B1-2							
DP-S1-2	2D12-1,11	P		E505			
DP-S1-2A	2D12-6	P		E509			
DP-S1-2D	2D12-7	P		E509			
BATT B2-1	1D21-1,4	P		E505			
DP-S2-1	1D21-3	P		E505			
DP-S2-1A	1M21A-80	P		E509			
H13-P621	1D11A-9	P	DP-S1-1A				
H22-P026	1D11A-8	P	DP-S1-1A				
SL-81	2D12D-2	C	DP-S1-2D	E509		877-5	DC Pwr Supply
SL-83	2D12D-3	C				877-5	
SM-8	2D12D-4	P				876-5	
C61-P001	2D12D-5	C				880-3	
RC-2	2P8AA-27	C	PP-8AA	508-1		886-5	
COHV-5B	2P8AA-36	C				885-4	
H13-P683	2P8AA-8	C					
H22-P017	2P8AA-6	C					
IR-22	2P8AG-1	C	PP-8A-G	508-3			
S2	2P8AG-3	C					
COHV2	2P8AF-4	C	PP-8A-F	508-1			

TABLE 040.75-8

EQUIPMENT	CABLE NO's	CA TYPE	POWER SOURCE	ELECTRICAL DRAWINGS ONE-LINE	WIRING	BLOCK	REMARKS
DG-2	2DG2-10	H		502-2			
SM-8	2SM8-110	H		502-2			
SL-81	2SM28-10	H		502-2			
	2SM8-40	H		502-2			
MC-8B	2SL81-10	P		503-8			
MC-8B-A	2M8B-10	P		503-7			
MC-8B-B	2M8B-20	P		503-12			
SL-83	2SM8-10	H		502-2			
MC-8F	2SL83-20	P		503-11			
MC-8A	2SL83-10	P		503-6			
PP-8A-A	2MBA-70	P		508-1			
PP-8A-F	2P8A-4	P		508-1			
PP-8A-B	2M8A-140	P		508-3			
PP-8A-G	2PBAB-2,50	P		508-3			
SL-81	2SL81-3	P		502-2		877-5	Gnd. Res.
	2SL8-75	C					
SL-83	2SL83-3	P		502-2		877-5	Gnd. Res.
	25	C					

TABLE 040.75-8

EQUIPMENT	CABLF NO's	CA TYPF	POWER SOURCE	ELECTRICAL DRAWINGS ONF-LINE	WIRING	BLOCK	REMARKS
Diesel Gen 2	2DG2-23	C				883-2	
	24	C					
	25	C					
	26	C					CT Leads
	31	C					
	36	C					
	41	C					
	44	C					
	45	C					
	47	C					
	50	C					
	2M12D-10	P	MC-S1-2D			883-2	
	11	C					
	20	P	MC-S1-2D				
	21	C					
	30	P	MC-S1-2D				
	31	C					
	40	P	MC-S1-2D				
	41	C					
	2SM28-11	C				883-2	
	14	C					
	2SM8-211						

TABLE 040.75-8

EQUIPMENT	CABLE NO's	CA TYPE	POWER SOURCE	ELECTRICAL DRAWINGS ONE-LINE	WIRING	BLOCK	REMARKS
Diesel Gen. 2	2SM8-35	C				883-2	
	115	C					
	116	C					
	208	C					
	209	C					
	210	C					
	BSL83-9047	C				883-2	Cont. ROA-FN-1B
	BSM3-9019	C				883-2	
	BSM8-9209	C				883-2	
	9240	C					
	9241	C					

TABLE 040.75-8

NOTES

1. Cables listed are those required for operation of equipment and instrumentation identified in Burns & Roe technical memorandum No. 1227.

2. Cables listed are external to PGCC.

3. Miscellaneous independent circuits related to dedicated shutdown equipment but not required for equipment power or control operation are not listed. Examples of such circuits are motor and equipment heater circuits, alarm and annunciator circuits, large motor ammeter circuits, etc.

All dedicated shutdown equipment is assumed to be in a mode compatible with plant power operation at the time the postulated fire occurs. No equipment is postulated to be in a test position at the time of the event.

5. Cable types indicate cable function as follows:

   P - Power
   C - Control and instrumentation

   Simple circuits having combined power and control functions (e.g. Solenoids) are listed as control circuits.

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TABLE 040.75-9

ASSIGNMENT OF EQUIPMENT OF DIVISION SEPARATION

DIVISION 1	DIVISION 2	DIVISION 3
RHR 2A	RHR 2B	HPCS 1
LPCS	RHR 2C	HPCS Diesel-Generator
Outboard Isolation Valves	Inboard Insolation Valves	125 Volt HPCS Battery
Standby Emergency Diesel Generator No. 1	Standby Emergency Diesel Generator No. 2	Power and Control for above
Standby Service Water Pump 1A	Standby Service Water Pump 1B	HPCS Standby Service Water Pump
RCIC		
Automatic Depressuriz-ation Division 1 Controls	Automatic Depressuriz-ation Division 2 Controls	Diesel Generating and switchgear HVAC System (Division 3)
Standby Gas Treatment (Division 1)	Standby Gas Treatment (Division 2)	Power and Control for above
Standby Liquid Control 1A	Standby Liquid Control 1B	
Reacor Closed Cooling Pump A	Reactor Closed Cooling Pump B & C	
Drywell Cooling Units A	Drywell Cooling Units B	
250 volt Battery 1		
125 volt Battery 1	125 volt Battery 2	
±24 volt Battery 1	±24 volt Battery 2	
Containment Atmosphere Control System (Division 1)	Containment Atmosphere Control System (Division 2)	
MSIV Leakage Control System (Inboard System)	MSIV Leakage Control System (Outboard System)	

TABLE 040.75-9

ASSIGNMENT OF EQUIPMENT OF DIVISION SEPARATION

DIVISION 1	DIVISION 2
Control Room, Cable Spreading, Diesel-Generating and Switchgear HVAC System (Division 1)	Control Room, Cable Spreading, Diesel-Generating and Switchgear HVAC System (Division 2)

DIVISION 4	DIVISION 5	DIVISION 6	DIVISION 7
RPS - A1	RPS - A2	RPS - B1	RPS - B2

TABLE 040.75-10

LIST OF FIRE AREAS

RB	-	REACTOR BLDG
RW	-	RADWASTE AND CONTROL BLDG
DG	-	DIESEL GENERATOR BLDG
TG	-	TURBINE GENERATOR BLDG
SB	-	SERVICE BLDG
SW1A	-	SERVICE WATER PUMPHOUSE 1A
SW1B	-	SERVICE WATER PUMPHOUSE 1B
CW	-	CIRCULATING WATER PUMPHOUSE
CT1	-	COOLING TOWER ELEC. BLDG. NO. 1
CT2	-	COOLING TOWER ELEC. BLDG. NO. 2
MWPH	-	MAKEUP WATER PUMPHOUSE
CST-TY	-	CONDENSATE STORAGE TANK AREA AND SWITCHYARD

TABLE 040.75-10

AREA IDENTIFICATION	BLDG	DESCRIPTION	ELEVATION	COMMENTS
R-I	RB	GEN EQPT AREA & STAIR S-3	422', 441', 471', 501', 522', 548', 572', 606'	
R-II	RB	PRIMARY CONTAINMENT	422', 441', 471', 499', 501', 519', 522', 541', 548', 567', 572', 606'	
R-III	RB	HPCS PUMP RM	422', 441'	
R-IV	RB	RHR HT EXCH B & PUMP RM R-1	422', 441', 471', 501', 522', 548', 572'	
R-V	RB	RHR HT EXCH A & PUMP RM R-2	422', 441', 471', 501', 522', 548', 572'	
R-VI	RB	RCIC PUMP RM R-3	422', 441'	
R-VII	RB	RHR PUMP RM R-4	422', 441'	
R-VIII	RB	LPCS PUMP RM R-5	422', 441'	
R-IX	RB	STAIR A-6	422', 441', 471', 501', 522', 548', 572', 606'	
R-X	RB	ELEV #3	422', 441', 471', 501', 522', 548', 572', 606'	
R-XI	RB	STAIR A-5	422', 441', 471', 501', 522', 548', 572', 606'	
R-XII	RB	ELEV #1	422', 441', 471', 501', 522', 548', 572', 606'	
R-XIII	RB	NOT USED	-	
R-XIV	RB	FUEL POOL COOL HX RM A	548'	
R-XV	RB	LOBBY FOR STAIR A-5 & ELEV #1	422'	
R-XVI	RB	NOT USED	-	
R-XVII	RB	SOUTH VALVE RM	471'	
R-XVIII	RB	DIV. 2 MCC RM	522'	
R-XIX	RB	H2 RECOMB CONT RM DIV. 2	572'	
R-XX	RB	DIV.1 MCC RM	471'	

TABLE 040.75-10

AREA IDENTIFICATION	BLDG	DESCRIPTION	ELEVATION	COMMENTS
RC-I	RW	GEN AREA	437', 452', 466', 467', 477', 487', 501'	
RC-II	RW	CABLE SPREADING RM	484'	
RC-III	RW	CABLE CHASE	467', 484', 487', 501', 507', 525'	
RC-IV	RW	ELEC. EQPT. RM. NO. 1	467'	INCLUDES BATT CHGR RM #1 & RPS RM #1
RC-V	RW	BATT RM #1	467'	INCLUDES BATT RM #1 & ELEC REPAIR SP.
RC-VI	RW	BATT RM #2	467'	
RC-VII	RW	ELEC. EQPT. RM. NO. 2	467'	INCLUDES BATT CHGR RM #2 & RPS RM #2
RC-VIII	RW	SWGR RM #2	467'	
RC-IX	RW	REMOTE SHUTDOWN RM	467'	
RC-X	RW	MAIN CONTROL RM	501'	
RC-XI	RW	HVAC EQPT RM - UNIT A	525'	UNIT A AIR CONDITIONING RM
RC-XII	RW	HVAC EQPT RM - UNIT B	525'	UNIT B AIR CONDITIONING RM
RC-XIII	RW	HVAC CHASE, COMM RM, INSTR SHOP	484', 501', 525'	
RC-XIV	RW	SWGR RM. #1	467'	
RC-XV	RW	STAIR A-8	437', 452', 467', 487', 501'	
RC-XVI	RW	STAIR A-7	437', 452', 467', 487', 501', 525'	
RC-XVII	RW	ELEV. #4	437', 452', 467', 487', 501'	
RC-XVIII	RW	STAIR A-13	467', 484'	
RC-XIX	RW	CORRIDOR C-205	467'	

TABLE 040.75-10

AREA IDENTIFICATION	BLDG	DESCRIPTION	ELEVATION
DG-I	DG	HPCS DG RM	441', 455'
DG-II	DG	1A DG RM	441', 455'
DG-III	DG	1B DG RM	441', 455'
DG-IV	DG	1A DIESEL OIL TNK PMP RM	441'
DG-V	DG	1B DIESEL OIL TNK PMP RM	441'
DG-VI	DG	HPCS DIESEL OIL TNK PMP RM	441'
DG-VII	DG	HPCS DIESEL OIL DAY TNK RM	441'
DG-VIII	DG	1A DIESEL OIL DAY TNK RM	441'
DG-IX	DG	1B DIESEL OIL DAY TNK RM	441'
DG-X	DG	FIRE DELUGE EQPT. RM	455'

TABLE 040.75-10

AREA IDENTIFICATION	BLDG	DESCRIPTION	ELEVATION	COMMENTS
TG-I	TG	GEN AREA	441', 471', 501'	*
TG-II	TG	TURB OIL STOR RM	441'	INCORP INTO TG-I
TG-III	TG	STAIR A-1	441', 471', 501'	
TG-IV	TG	ELEV #3	441', 471', 501'	
TG-V	TG	AUX BOILER RM	441'	INCORP INTO TG-I
TG-VI	TG	STAIR A-3	441', 471', 501'	
TG-VII	TG	H2 SEAL OIL RM	441'	INCORP INTO TG-I
TG-VIII	TG	STAIR A-4	441', 471', 501'	
TG-IX	TG	TURB OIL RES RM	471'	INCORP INTO TG-I
TG-X	TG	TRANSFORMER RM	441'	INCORP INTO TG-I

*Area TG-I: Includes Main Plant Corrisdors and Reactor Bldg. Main Steam Tunnel

TABLE 040.75-10

AREA IDENTIFICATION	BLDG	DESCRIPTION	ELEVATION	COMMENTS
S-I	SB	GEN AREA	420', 441', 456'	INCLUDES S-II,-III,-IV,-V,-VI,-VII
S-II	SB	CHLORINE RM	420'	INCORP INTO S-I
S-III	SB	STAIRWAY A-10	420', 441'	INCORP INTO S-I
S-IV	SB	STAIRWAY A-9	420', 441'	INCORP INTO S-I
S-V	SB	STAIRWAY A-14	441', 456'	INCORP INTO S-I
S-VI	SB	QUAL ASSUR VAULT	456'	INCORP INTO S-I
S-VII	SB	HVAC RM	456'	INCORP INTO S-I

TABLE 040.75-10

AREA IDENTIFICATION	BLDG	DESCRIPTION	ELEVATION	COMMENTS
SW-I	SWIA	PUMP AREA	431', 441'	
SW-II	SWIB	PUMP AREA	431', 441'	
CST-TY	-	COND STOR TNK BASIN AND SWITCHYARD	441'	NOT ADDRESSED IN FINAL REPORT
CW-I	CWPH	DIESEL FUEL STORAGE	ABOVE 448'	
CW-II	CWPH	PUMP AREA	ABOVE 448', BELOW 448'	
CT-I	COOL TWR	GEN. AREA	448'	
CT-II	ELEC BLDGS	GEN. AREA	448'	
MWPH	MAKEUP WTR PUMPHOUSE	GEN. AREA	375'	
WW-II	WELL WATER PUMPHOUSE NO. 2	SEPARATE PUMPHOUSE IN OUTDOOR AREA		

Q. 040.076

Provide a table that lists Class 1E and non-Class 1E cables that are associated with the essential safe shutdown systems identified in Q. 040.075. For each cable listed (see note below):

a. Define the cables' association to the safe shutdown system (common power source, common raceway, separation less than the IEEE Standard 384 quidelines, cables for equipment whose spurious operation will adversely affect shutdown systems, etc.).

b. Describe each associated cable routing (by fire area) from source to termination; and,

c. Identify each location where the associated cables are separated by less than a wall having a three-hour fire rating from cables required for or associated with any redundant shutdown system.

NOTE: Option (a) of Q. 040.077 is considered to be one method of meeting the requirements of Section III.G.3 Appendix R. If option (a) is selected, the information requested in items (a) and (c) above should be provided in general terms and the information requested by (b) above need not be provided.

Response

Circuits referred to in Appendix R as being associated with essential safe shutdown systems are referred to as "Allied Circuits" on WNP-2. The choice of the term "Allied Circuits" was arbitrary, and was made for the sole purpose of facilitating a fire hazards shutdown analysis free of misinterpretation. The term "Associated" is reserved for use in conjunction with circuits defined as "Associated" under the guidelines of Regulatory Guide 1.75.

The manner in which Allied Circuits impact the fire hazard shutdown analysis for any specific fire area is dependent upon the type of analysis being utilized for the particular fire area (Refer to Question 040.075 for a discussion of fire hazard shutdown analysis methodology).

For the case of a fire area analyzed by means of the Redundant Fire Area method, direct consideration of all Allied Circuits is not required. Analysis for divisional fire areas is dependent only upon review of the impact of fire upon intruding Class 1E and Associated Circuits. In fact, all Associated Circuits of the electrical separation divisions which are not compatible with the fire area divisional assignment are the Allied Circuits, although they are not referred to as such. Analysis of only the intruding

Associated Circuits is required for the Fire Hazards Shutdown Analysis contained in Appendix F, with the exception of the analysis of the potential for "bridging" of Associated Circuits across the plant electrical separation divisions. Analysis of the "bridging" circuit condition is presently being addressed as part of an overall electrical separation review for WNP-2 (Reference: B&R Richland Office, BOP Separation Study - Task 3670). This subject is being addressed and resolved with NRC Offices separately from the Appendix F (Reference: NRC to WPPSS letter, "Staff Response to the Presently Proposed Cable Separation Criteria for the WNP-2 Facility", dated May 4, 1981 and GO2-81-146).

For the case of a fire area analyzed by means of the Dedicated Shutdown System method, consideration of all Allied Circuits is required. A listing of Allied Circuits will be provided in the Fire Hazards Shutdown Analysis to be included in Appendix F. That listing will indicate whether the Allied Circuit is a Class 1E, Associated or non-Class 1E circuit. Analysis will be provided.

a. A review is presently in progress to determine all Dedicated Shutdown System Allied Circuits allied by virtue of common power supply. Allied Circuits of this type are common to all fire areas which utilize the dedicated shutdown method. These circuits will be identified, reviewed for impact on the dedicated shutdown system, and modified if required. Data and results will be incorporated into Appendix F.

   A review to determine all Dedicated Shutdown System Allied Circuits allied by virtue of location and routing cannot be completed until a final determination is made regarding the exact method of protecting dedicated shutdown cables for the particular fire area involved. Rerouting of dedicated cables, for example, could affect the Allied Circuit listing for the fire area. Circuits allied by virtue of location and routing will be unique for each area utilizing the dedicated shutdown method. A review of these circuits will be provided later; data and results will be incorporated into Appendix F.

b. Source to termination cable routing data will not be provided for intruding Associated Circuits (Redundant Fire Area method) or Allied Circuits (Dedicated Shutdown method). However, analysis performed in accordance with the Dedicated Shutdown method will address the following:

   1. Common Power Supply: Question 040.075, Table 040.75-7 identifies all electrical distribution system components which are power sources for dedicated shutdown system equipment. Table 040.75-8 identifies the dedicated shutdown cables receiving power from those sources. Any other cables powered from those sources are, therefore, allied. These circuits can be checked at the source to determine their potential for impact on dedicated shutdown circuits, and corrective measures taken as required.

2. Location and Routing: For circuits allied by virtue of locating and routing, the circuit source will be identified and reviewed to assure that fire initiated impacts on the Allied Circuit cannot impact the dedicated shutdown circuits. The fire area location of the circuit power source will be provided to assure that any protective electrical components (fuses, etc.) upon which analysis depends are not located in the same fire area as the fire itself.

c. As indicated above, Allied Circuits need not be considered directly for fire areas analyzed via the Redundant Area method.

The dedicated shutdown analysis is utilized only in general fire areas where there are significant amounts of cabling of more than one electrical separation division. Since virtually all dedicated shutdown circuits are Class 1E, an Allied Circuit can only be allied with and contain the potential for impact upon one of the safety related divisions in the area (Refer to electrical separation criteria contained in Chapter 8.3, and "bridging" analysis discussed above, for verification).

Therefore, in every general fire area in which Allied Circuits exist, Allied Circuits are not separated by a wall having a three-hour fire rating from cabling of a division that is redundant to the division with which they are allied. The separation provided is in accordance with the criteria contained in Chapter 8.3.

This situation could also exist in Divisional Fire Areas, although to a much lesser degree. Again, however, divisional cable separation in accordance with Chapter 8.3 is provided.

Provide one of the following for each of the circuits identified in Q. 040.076 item (c):

a. The results of an analysis that demonstrates that failure caused by open, ground, or hot short of cables will not affect its association shutdown system (see note below).

b. Identify each circuit requiring a solution in accordance with Section III.G.3 of Appendix R; or,

c. Identify each circuit meeting or that will be modified to meet the requirements of Section III.G.3 of Appendix R (i.e., three-hour wall, 20 feet of clear space with automatic fire suppression, or one-hour barrier with automatic fire suppression).

NOTE: Option (a) above is considered to be one method of meeting the requirements of Section III.G.3 Appendix R. If option (a) above is selected, the information requested in Q. 040.076 items (a) and (c) should be provided in general terms and the information requested by Q. 040.076 item (b) need not be provided.

Response

The method utilized for the Fire Hazard Analysis of a particular fire area determines the method and extent to which Allied Circuits are analyzed.

As indicated in Question 040.076, all Associated Circuits that are not compatible with the divisional assignment of a divisional fire area are Allied Circuits. This is a result of the assumption that all Class 1E cabling, except that corresponding to the fire area divisional assignment, is required for shutdown, unless otherwise justified (intruding cables). However, the only Allied Circuits of concern are those Associated Circuits located in the fire area undergoing analysis. These intruding circuits are all individually analyzed to determine the effects their fire related failures might have on the Class 1E systems, and the acceptability/unacceptability of those effects.

As indicated in Question 040.076, all circuits allied with the Dedicated Shutdown System undergo analysis.

a. The results of all analysis will be provided in the Fire Hazards Shutdown Analysis contained in Appendix F.

The effects of opens, shorts, grounds and hot shorts upon the load side portions of intruding Associated Circuits (Redundant Area method) or Allied Circuits (Dedicated Shutdown method) is of no consequence. Neither analytical method depends upon functionality of this equipment to achieve shutdown. Malfunction of the load side components cannot defeat the safe shutdown systems.

The upstream effects of open intruding Associated Circuits (Redundant Area method) on the shutdown systems do not require analysis. If the circuit was of a type that a simple open could defeat functionality of Class 1E equipment, it would have been designated as a Class 1E circuit, not as an Associated Circuit. Specific review of the upstream effects of open Allied Circuits (Dedicated Shutdown method) on the dedicated shutdown systems will be addressed as part of the analysis discussed in Question 040.076a.

The upstream effects of a shorted or grounded intruding Associated Circuit (Redundant Area method) do not require analysis. All Associated Circuits associated by virtue of power supply are provided with fuses or other devices to isolate them from the upstream Class 1E system and limit the effects of events in the downstream Associated Circuits from causing unacceptable consequences in the upstream Class 1E systems. Associated Circuits associated by virtue of location and routing can only cause impact to the upstream non-Class 1E systems; these systems are not utilized for shutdown. As a part of the analysis discussed in Question 040.076a., all Allied Circuits (Dedicated Shutdown method) will be reviewed to determine that upstream protective devices (fuses or other devices) are provided to prevent faulted circuits from causing impact to the dedicated shutdown system. For the case of circuits allied by virtue of location or routing, it will also be verified that the protective devices are not located in the same fire area in which the fire is occurring.

The above statements regarding shorted or grounded circuits applies to hot shorts also.

b,c. The analysis to be provided in Appendix F will identify all circuits requiring resolution in accordance with Appendix R requirements.

Q. 040.078

To assure compliance with GDC 19, we require the following information be provided for the control room. If credit is to be taken for an alternate or dedicated shutdown method for other fire areas (as identified by Q. 040.075 item (c) or Q. 040.077 item (b)) in accordance with Section III.G.3 of new Appendix R to 10CFR Part 50, the following information will also be required for each of these plant areas.

a. A table that lists all equipment including instrumentation and vital support system equipment that are required by the primary method of achieving and maintaining hot and/or cold shutdown.

b. A table that lists all equipment including instrumentation and vital support system equipment that are required by the alternate, dedicated, or remote method of achieving and maintaining hot and/or cold shutdown.

c. Identify each alternate shutdown equipment listed in item (b) above with essential cables (instrumentation, control and power) that are located in the fire area containing the primary shutdown equipment. For each equipment listed provide one of the following:

    1. Detailed electrical schematic drawings that show the essential cable that are duplicated elsewhere and are electrically isolated from the subject fire areas, or

    2. The results of an analysis that demonstrates that failure (open, ground, or hot short) of each cable identified will not affect the capability to achieve and maintain hot or cold shutdown.

d. Provide a table that lists Class 1E and non-Class 1E cables that are associated with the alternate, dedicated, or remote method of shutdown. For each item listed, identify each associated cable located in the fire area containing the primary shutdown equipment. For each cable so identified, provide the results of an analysis that demonstrates that failure (open, ground, or hot short) of the associated cable will not adversely affect the alternate, dedicated, or remote method of shutdown.

Response

See response to Question 040.075 for a general discussion of the methodology utilized in the Fire Hazard Shutdown Analysis.

a. Tables 040.075-1 thru 7 list all equipment required for operation of the Dedicated Shutdown System. This is the equipment required for achieving and maintaining hot and/or cold shutdown in the event of fires occurring in general fire areas. The fire area location of all equipment is indicated on the Tables.

No listing is provided for equipment required to achieve and maintain hot and/or cold shutdown in the event of a fire in divisional fire areas (Redundant Area method). This list would include every safety related component in the plant. This is a result of the analysis methodology that given a fire in an area assigned to one safety related division, all equipment in the other safety related divisions is required, unless otherwise specifically justified (intruding cables and equipment).

b. The question of alternate, dedicated or remote methods of achieving and maintaining hot and/or cold shutdown does not apply to areas analyzed by means of the Redundant Area method (Divisional Fire Areas).

General fire areas analyzed by means of the Dedicated Shutdown method depend upon that dedicated shutdown system to achieve and maintain hot and/or cold shutdown. See item (a) response above. It is, therefore, the primary method of achieving shutdown; it is also the only analytically guaranteed method of achieving a fire related shutdown.

c. There is no alternate shutdown system utilized in the WNP-2 Fire Hazard Shutdown Analysis.

d. See response to Question 040.076 and 040.077.

The above provides response to this question's request for data regarding the shutdown systems for "other fire areas" (other than the Main Control Room).

The Main Control Room represents a unique case in terms of fire analysis. Dedicated shutdown components have the capability of control from either the Main Control Room or the Remote Shutdown Room. Transfer switches located in the Remote Shutdown Room transfer control to that location in the event of a catastrophic fire in the Main Control Room. Credit is taken for operator action to manually initiate reactor SCRAM and MSIV closure prior to leaving the Main Control Room. Main Control Room fire analysis consists of verification that the effects of fire in the Main Control Room cannot disable dedicated shutdown component operability following transfer of plant control to the Remote Shutdown Room.

Basic circuit design and layout for nuclear/mechanical dedicated shutdown components (pumps, valves, etc.) routes cabling from remote devices to the Remote Shutdown Room boards C61-P001 (NSSS) and RS (BOP) for control at that location. Circuitry is then extended into the Main Control Room for control from that location also. For normal plant operation, the Remote Shutdown Room transfer switches remove the Remote Shutdown Room control switches from the circuits (through open transfer switch contacts) and insert the Main Control Room control switches (through closed transfer switch contacts). To transfer dedicated shutdown component control to the Remote Shutdown Room, the transfer switch contact positions reverse.

Initial analysis of the capability to achieve cold reactor shutdown in the event of a Main Control Room fire assumes that:

a. The entire Main Control Room is lost (not considered credible).

b. Operators SCRAM the reactor and initiate MSIV closure as they evacuate the area.

Analysis then verifies the following:

a. Dedicated shutdown component cabling from remote locations does not route to the Remote Shutdown Room via the Main Control Room.

b. Dedicated shutdown component circuitry contains the previously described transfer arrangement (pumps, valves, etc.)

c. Dedicated shutdown components which do not contain the previously described transfer arrangement (fans, diesel generators, etc.) can be controlled from an alternate remote location, and are not disabled by the effects of Main Control Room fire.

The controls and displays are listed in FSAR Section 7.4.1.4.

The residual heat removal system is generally a low pressure system that interfaces with the high pressure primary coolant system. To preclude a LOCA through this interface, we require compliance with the recommendations of Branch Technical Position RSB 5-1. Thus, this interface most likely consists of two redundant and independent motor operated valves with diverse interlocks in accordance with Branch Technical Position ICSB 3. These two motor operated valves and their associated cable may be subject to a single fire hazard. It is our concern that this single fire could cause the two valves to open resulting in a fire-initiated LOCA through the subject high-low pressure system interface. To assure that this interface and other high-low pressure interfaces are adequately protected from the effects of a single fire, we require the following information:

a. Identify each high-low pressure interface that uses redundant electrically controlled devices (such as two series motor operated valves) to isolate or preclude rupture of any primary coolant boundary.

b. Identify each device's essential cabling (power and control) and describe the cable routing (by fire area) from source to termination.

c. Identify each location where the identified cables are separated by less than a wall having a three-hour fire rating from cables for the redundant device.

d. For the areas identified in item (c) above (if any), provide the bases and justification as to the acceptability of the existing design or any proposed modifications.

Response

a. The major reactor pressure boundary high pressure/low pressure interface valves have been reviewed per Branch Technical Position RSB 5-1. Using these criteria, check valves in series with motor operated valves (MOV's) are acceptable. A fire could open only the MOV. Many occurrences of this combination of check and MOV exist in the Core Spray, Feedwater, and Residual Heat Removal Systems, among others. A pneumatic operator is usually associated with the check valve. This operator is for testing purposes only and can neither unseat nor prevent from seating the valve flapper when a differential pressure exists across the valve. Hence, a fire-caused failure of the solenoid actuators for the pneumatic operators on these check valves cannot cause the valves to open inadvertently and thus cannot degrade the reactor coolant pressure boundary.

In addition to the above, five pairs of valves all associated with the RHR System as high/low pressure interface valves, consist of two remotely operated valves in series. The first pair of these valves are in the shutdown cooling suction line. The second and third pair are in the lines of each RHR loop for shutdown return. The fourth and fifth pair are in the lines to each RHR heat exchanger for use in the steam condensing mode. The valve numbers are given in Table 040.79-1.

The shutdown cooling suction valves (RHR-V-8, 9) are in separate divisions and separate fire zones. The inboard (upstream) isolation valve (RHR-V-9) is in Div. 2 and is located in the inerted containment where a fire cannot be postulated. RHR-V-8 is in Div. 1 located outside containment.

The upstream shutdown cooling return valves (RHR-V-53A, B) are in the same division (Div. 1), but separated by the diameter of the containment vessel. The inboard (downstream) by-pass valves (RHR-V-123A, B) are located in the inerted containment where a fire cannot be postulated. They are both in Div. 2.

The steam condensing mode valves constitute a high-low pressure interface between the main steam and the RHR piping. For RHR Loop A should both motor operated valves be driven open, i.e. RHR-V-52A and RHR-V-87A, adequate over protection exists via relief valves RHR-RV-55A and RHR-RV-95A to prevent rupture of the downstream RHR piping. For loop B, the same protection exists via RHR-RV-55B and RHR-RV-95B for the RHR piping downstream of condensing mode motor operated valves RHR-V-52B and RHR-V-87B should they both be driven open.

Isolation upstream of valves RHR-V-52A and B is provided by the RCIC steam isolation valves RCIC-V-64 (Div. 1) and RCIC-V-63 (Div. 2), and the 1" by-pass RCIC-V-76 (Div. 2). Should a fire open RHR-V-52A and RHR-V-87A or RHR-V-52B and RHR-V-87B, a plant operator would close the inboard RCIC valves, isolating the RHR valves from the reactor pressure boundry. The requirement for this action by a plant operator is deemed highly unlikely since the outboard isolation valve RCIC-V-64 is normally maintained in a closed position.

b. Identification of power and control cabling for the valves listed in Table 040.79-1, as well as data regarding the cable routing, is provided in the following tables:

Valve Pair	Table
RHR-V-8/RHR-V-9	040.79-2
RHR-V-52A/RHR-V-87A	040.79-3
RHR-V-52B/RHR-V-87B	040.79-4
RHR-V-53A/RHR-V-123A	040.79-5
RHR-V-53B/RHR-V-123B	040.79-6

Cabling for circuits which enter the Main Control Room and do not wire directly to a benchboard or vertical panel assembly is tracked only as far as the termination cabinet within the Main Control Room.

c. Review of Table 040.79-2 thru 6 indicates that cabling for both valves (of any valve pair) routes through the following common fire areas:

Valve Pair	Common Fire Areas
RHR-V-8/9	R-I, RC-II, RC-III, RC-IX, RC-X
RHR-V-52A/87A	R-I, RC-II, RC-III, RC-X
RHR-V-52B/87B	R-I, R-IV, R-XIX, RC-II, RC-III, RC-IX, RC-X
RHR-V-53A/123A	R-I, RC-III, RC-X
RHR-V-53B/123B	R-I, RC-II, RC-III, RC-X

d. The availability of full flow relief valves discussed in item a. precludes any problem with high pressure-low pressure interface valves RHR-V-52A/87A and RHR-V-52B/87B.

Protection of one of the two valves of any pair of the remaining pairs of valves listed in Table 040.79-1 is required to assure that both valves from any one pair cannot concurrently open. Protection of cabling for valves RHR-V-9/123A/123B will be included in plant revisions determined to be required as a result of the 10CFR50, Appendix R Fire Hazards shutdown analysis.

Protection of power cabling is not required since the valve is not required to change position.

Protection of control cabling will address the following cables:

Fire area R-I: RHR-V-9: 2M8BA-312,-314,-316
RHR-V-123A: 2M8BA-502,-503
RHR-V-123B: 2M8BA-432,434
Fire area RC-II: RHR-V-9: 2M8BA-314,-315,-316
RHR-V-123A: 2M8BA-502
RHR-V-123B: 2M8BA-434
Fire area RC-III: RHR-V-9: 2M8BA-314,-316
RHR-V-123A: 2M8BA-502
RHR-V-123B: 2M8BA-434
Fire areas RC-IX, RC-X: RHR-V-9, RHR-V-123A, and RHR-V-123B

Cabling terminates in these areas. Resolution is presently being discussed with General Electric Co. and will be provided in a subsequent amendment.

TABLE 040.79-1

HIGH-LOW PRESSURE INTERFACE WITH REDUNDANT ELECTRICALLY CONTROLLED DEVICES

DESCRIPTION	UPSTREAM COMPONENT	DOWNSTREAM COMPONENT
Steam Condensing	RHR-V-52A (Motor operated 8" globe)	RHR-V-87A* (Motor operated 8" globe)
	RHR-V-52B (Motor operated 8" globe)	RHR-V-87B* (Motor operated 8" globe)
Shutdown Cooling (Suction)	RHR-V-9 (Motor operated 20" gate)	RHR-V-8 (Motor operated 20" gate)
Shutdown Cooling (Return)	RHR-V-53A (Motor operated 12" globe)	RHR-V-123A (Motor operated 1" gate by-pass around 12" check valve RHR-V-50A)
	RHR-V-53B (Motor operated 12" globe)	RHR-V-123B (Motor operated 1" gate by-pass around 12" check valve RHR-V-50B)

* Alternate downstream components. RHR-FCV-51A and RHR-FCV-51B are electrically controlled, air operated valves.

TABLE 040.79-2

VALVE DATA		CABLE DATA				
Valve	Description	Cable No.	Type	Function	Fire Areas	Raceway
RHR-V-8 (E12-F008)	125 Vdc Div. I RHR Suction Cooling Valve	1M21A-20	P	Power Supply: MC-S2-1A, Part B to valve	R-XX, R-I	T&C
		1M21A-21	C	Control: MC-S2-1A, Part B to pos. sw. on valve	R-XX, R-I	T&C
		1M21A-22	C	Control: MC-S2-1A, Part B to Remote Shutdown Pnl.C61-P001	R-XX, R-I, RC-III, RC-II, RC-IX	T
		1M21A-23	C	Control: Remote Shutdn. Pnl.C61-P001 to Term.Cab. H13-P684G Mod. 13	RC-IX, RC-II, RC-X	T
		1M21A-24	C	Control: MC-S2-1A, Part B to Remote Shutdn. Pnl. C61-P001	R-XX, R-I, RC-III, RC-II, RC-IX	T
		AIVD-9093	C	Indic: Valve Position Indication TB-R370 to pos.sw.	R-I	T&C
		1RHR-20	C	Control: Limit sw. Int'lk to E12-C002A(RHR-P-2A) stop ckt. Cont. Rm. Term Cab. H13-P684F Mod.24 to valve E12-F008 operator	RC-X, RC-II, RC-III, R-I	T&C
RHR-V-9 (E12-F009)	480 Vac. Div. II RHR Suction Cooling Valve	2M8BA-310	P	Power Supply: MC-8B-A to Penetration Assembly	R-XVIII, R-I	T
		2M8BA-311	P	Power Supply: Penetration Assembly to valve	R-II	C
		2M8BA-312	C	Control & Indic.: MC-8B-A to Penetration Assembly	R-XVIII, R-I	T
		2M8BA-313	C	Control & Indic.: Penetration Assembly to valve	R-II	C
		2M8BA-314	C	Control & Indic.: MC-8B-A to to Remote Shutdn. Pnl. C61-P001	R-XVIII, R-I RC-III, RC-II, RC-IX	T
		2M8BA-315	C	Control & Indic.: Rem. Shutdn. Pnl. C61-P001 to Term. Cab. H13-P683 Mod. 42	RC-IX, RC-II,. RC-X	T

TABLE 040.79-2 (continued)

VALVE DATA		CABLE DATA				
Valve	Description	Cable No.	Type	Function	Fire Areas	Raceway
RHR-V-9 (Cont'd)	-	2M8BA-316	C	Indication: MC-8B-A Remote Shutdn. Pnl. C61-P001	R-XVIII, R-I, RC-III, RC-II, RC-IX	T
		2RHR-19	C	Pump E12-C002B valve stop control. TB-C513 to valve operator	R-II	C
		2RHR-47	C	Pump E12-C002B valve stop control. Cont. Rm. to Penetration Assembly.	RC-X, RC-II, RC-III, R-I	T
		BIVD-9143	C	Indication: TB-C533 to pos.sw. valve RHR-V-9	RC-II	C
		BIVD-9160	C	Indication: TB-C523 to TB-C533	R-II	C
		BIVD-9087	C	Indication: Vert. Bd. "S" to TB-R323	RC-X, RC-II, RC-III, R-I	T

P-Power Supply Cable
C-Control Cable
T-Tray
C-Conduit

TABLE 040.79-3

VALVE DATA		CABLE DATA				
Valve	**Description**	**Cable No.**	**Type**	**Function**	**Fire Areas**	**Raceway**
RHR-V-52A (E12-F052A)	480 Vac. Div. I RHR Isolation Valve	1M7BB-190	P	Pwr. Supply MC-7B-B to valve	R-I	T&C
		1M7BB-191	C	Control MC-7B-B to pos. sw. on valve	R-I	T&C
		1M7BB-192	C	Control & Ind.: from Term. Cab. H13-P684B to MC-7B-B	R-I, RC-III, RC-II, RC-X	T
RHR-V-87A (E12-F087A)	480 Vac. Div. I RHR Steam Line Isolation Valve	1M7BB-200	P	Pwr. supply MC-7B-B to valve	R-I	T&C
		1M7BB-201	C	Control MC-7B-B to valve	R-I	T&C
		1M7BB-202	C	Control & Ind.: from Term. Cab. H13-P684B to MC-7B-B	R-I, RC-III, RC-II, RC-X	T

P-Power Supply Cable
C-Control Cable
T-Tray
C-Conduit

TABLE 040.79-4

VALVE DATA		CABLE DATA				
Valve	Description	Cable No.	Type	Function	Fire Areas	Raceway
RHR-V-52B	480 Vac. Div. II RHR Isolation Valve	2M8BB-250	P	Power Supply: MC-8B-B to valve	R-XIX, R-I, R-IV	T&C
		2M8BB-251	C	Control: MC-8B-B to valve	R-XIX, R-I, R-IV	T&C
		2M8BB-252	C	Control: MC-8B-B to Remote Shutdn. Pnl. C61-P001	R-XIX, R-I, RC-III, RC-II, RC-IX	T
		2M8BB-253	C	Control: Remote Shutdn. Pnl. C61-P001 to Term Cab. H13-P683B	RC-IX, RC-II, RC-X	T
RHR-V-87B	480 Vac. Div. II RHR Steam Line Valve	2M8BB-210	P	Power Supply: MC-8B-B to valve	R-XIX, R-I R-IV	T&C
		2M8BB-211	C	Control: MC-8B-B to valve	R-XIX, R-I, R-IV	T&C
		2M8BB-212	C	Control: MC-8B-B to Remote Shutdn. Pnl. C61-P001	R-XIX, R-I, RC-III, RC-II, RC-IX	T
		2M8BB-213	C	Control: Remote Shutdn. Pnl. C61-P001 to Term. Cab. H13-P683B	RC-IX, RC-II, RC-X	T

P-Power Supply Cable
C-Control Cable
T-Tray
C-Conduit

TABLE 040.79-5

VALVE DATA		CABLE DATA				
Valve	Description	Cable No.	Type	Function	Fire Areas	Raceway
RHR-V-53A (E12-F053A)	480 Vac. Div. I RHR Injection Valve	1M7BA-130	P	Power Supply: MC-7B-A to valve	R-I	T&C
		1M7BA-132	C	Control: MC-7B-A to valve	R-I	T&C
		1M7BA-133	C	Control: MC-7B-A to Term. Cab. H13-P684B	R-I, RC-III, RC-II, RC-X	T
RHR-V-123A (E12-F099A)	480 Vac. Div. II RHR Valve	2M8BA-500	P	Power Supply: for vlv. MC-8B-A to TB-R321	R-XVIII, R-I	T
		2M8BA-501	P	Power Supply: TB-C521 to valve	R-II	C
		2M8BA-502	C	Control: MC-8B-A to Term. Cab. H13-P683F	R-XVIII, R-I, RC-III, RC-II, RC-X	T
		2M8BA-503	C	Control: MC-8B-A to TB-R313	R-XVIII, R-I	T
		2M8BA-504	C	Control: TB-C513 to valve	R-II	C
		2M8BA-507	C	Term. Cab. K2 to Relay Cab. RC-2	RC-X	C

P-Power Supply Cable
C-Control Cable
T-Tray
C-Conduit

TABLE 040.79-6

VALVE DATA		CABLE DATA				
Valve	Description	Cable No.	Type	Function	Fire Areas	Raceway
RHR-V-53B (E12-F053B)	480 Vac. Div. I RHR Injection Valve	1M7BA-580	P	Power Supply: MC-7B-A to valve	R-I	T&C
		1M7BA-581	C	Control: MC-7B-A to pos.sw. on valve	R-I	T&C
		1M7BA-582	C	Control: MC-7B-A to Rem. Shutdn. Pnl. C61-P001	R-I, RC-III, RC-II, RC-IX	T
		1M7BA-583		Control: Remote Shutdn. Pnl. C61-P001 to Term. Cab. H13-P684B	RC-IX, RC-II, RC-X	T
RHR-V-123B (E12-F099B)	480 Vac. Div. II RHR Valve	2M8BA-430	P	Power Supply: MC-8B-A to TB-R321 for valve	R-XVIII, R-I	T
		2M8BA-431	P	Power Supply: TB-C521 to valve	R-II	C
		2M8BA-432	C	Control: MC-8B-A to TB-R313 for valve	R-XVIII, R-I	T
		2M8BA-433	C	Control: TB C513 to pos. sw. on valve	R-II	C
		2M8BA-434	C	Control: MC-8B-A to Term. Cab. H13-P683F	R-XVIII, R-I, RC-III, RC-II RC-X	T

P-Power Supply Cable
C-Control Cable
T-Tray
C-Conduit

Q. 010.035
(3.4.1)

The FSAR states that the "Seismic Category I piping and electrical conduit penetrations that are below grade ... are ... not sealed against groundwater pressure." Demonstrate that the safety functions would not be compromised by water flowing into the building through these piping and conduit penetrations as the result of the following events:

a. Another compartment is flooded and water is flowing out of the building through the piping and conduit penetrations, resulting in saturated ground conditions.

b. A non-Seismic Category I tank ruptures emptying all of its contents.

Response:

Please refer to revised 3.4.1.4.2, page 3.4-4 for the information requested.*

The design and installation of the boots at pipe penetrations and of the silicon foam at conduit penetrations, described in 3.4.1.4.2, provide waterproof penetrations that are capable of preventing the compromise of safety functions by events such as those stated in the question.

The response to Question 010.010 has been similarly revised.*

*Draft FSAR page changes attached.

REPLACE WITH ATTACHED INSERT (4)

The lowest floor surface in the reactor building is the top of the foundation mat at elevation 422'-3" MSL. Since this is above the design basis groundwater level, the structure will be unaffected by the force effects of bucyancy and static water due to groundwater at elevation 420 feet MSL.

Groundwater elevation 420 feet MSL has been compared with foundation levels of Seismic Category I structures and it has been found that waterproofing is not required.

Seismic Category I piping and electrical conduit penetrations that are below grade are above the design basis groundwater table and are therefore not sealed against groundwater pressure.

The only materials underlying the site that might exhibit unfavorable response to seismic or other events under saturated soil conditions are the loose to medium dense, fine to coarse sand with scattered gravel, in the upper approximate 40 feet of the soil profile. These are removed and recompacted as structural fill, as described in 2.5.4.8 and 2.5.4.12. Structural fill supports the Seismic Category I structures, including the turbine generator building and service building, in the central plant complex. Structural fill, as required, is also utilized below the other Seismic Category I and safety-related structures including underground piping and electrical duct banks. The structural fill is compacted to a minimum of 75% relative density and an average relative density of not less than 85%.

To evaluate the possible effects of a gross rise in groundwater levels, Shannon and Wilson, soil consultants, performed a series of repetitive triaxial tests in identical soils in the dry and saturated states and concluded that saturation would not necessitate changes in allowable bearing pressures or settlement calculations as discussed in 2.5.4.10.

To provide conservatism in the design of structures, the seismic dynamic reponse of the structures and components is examined over a range of soil shear moduli, as discussed in 2.5.4.7.

According to 3.6, Groundwater, of the Shannon and Wilson Supplementary Soil Investigation in Appendix 2.5F, the compacted backfill discussed above will eliminate

Insert to Page 3.4-4:

Seismic Category I piping and electric conduit penetrations that are below grade are above the design basis groundwater table, and sealing against groundwater pressure is therefore not required. However, all pipes penetrating exterior walls are water-proofed sealed by boots installed on both sides of the wall penetration; all electrical conduit penetrations are through-wall sealed using silicon foam, also a waterproof sealer.

Q 010.10
(3.4.1)

Demonstrate that all piping and electrical penetrations in safety-related structures that are below the level of the Probable Maximum Flood are water-tight.

Response:

As stated in 3.4.1.4.1 the plant site grade is higher than the design basis flood elevation resulting from the probable maximum precipitation (PMP) event. Due to the short duration of the PMP flood, the ground water level at the plant site is not affected. As stated in 3.4.1.4.2, piping and electrical penetrations are above the design basis groundwater level and ~~was~~ therefore ~~not sealed~~ sealing against groundwater pressure is not required. However, all pipes penetrating exterior walls are water-proofed sealed by boots installed on both sides of the wall penetration; all electrical conduit penetrations are through-wall sealed using silicon foam, also a waterproof sealer.

Q. 010.036
(3.5.1)
RSP

It is the Staff's position that all safety-related equipment shall be appropriately protected against the effects of internally generated missiles in accordance with Title 10, Code of Federal Regulations Part 50, Appendix A, General Design Criteria 4. The effects of internally generated missiles such as valves stems, bonnets, control rod drive mechanisms, and high pressure accumulators impacting onto safety-related equipment must be evaluated. Appropriate protection must be provided to assure that a missile will not prevent a safe shutdown of the plant or result in uncontrolled release of radioactivity during normal operation or during the most severe design basis accident with the most limiting single active failure. Describe the means provided for assuring protection of safety-related equipment from all internally generated missiles.

Response:

The preferred method of protecting safety-related equipment from internally generated missiles is by separation and redundancy. In addition, potential missile sources have been oriented away from essential equipment wherever practical.

A pipe break/missile study evaluation is currently underway at WNP-2. This study will consider the effects of postulated valve stem, bonnet, control rod drive mechanism, and high pressure accumulator missiles. Appropriate protection will be provided where necessary.

Preliminary results scheduled for submittal in the last quarter of 1981 will address effects inside of containment and will also delineate methodology used in the study. A follow-up submittal will identify effects on the remaining areas outside of containment. Upon completion, the study will provide assurance that essential systems have been adequately protected from internally generated missiles.

Q. 010.037
(3.5.1.2)

Regulatory Guide 1.70, Revision 3, Section 3.5.1.2, requires that the structures, systems, and components protected by physical barriers should be identified. The discussion and the figures in the FSAR do not indicate where, if at all, physical missile barriers are used.

Identify all structures, systems, and components that are protected by physical barriers. Provide a description of the types of physical barriers that are employed at your plant.

Response:

At WNP-2, the preferred method of protecting essential systems and components from internally generated missils is by separation and redundancy. In addition, potential missile generating sources have been oriented away from essential systems and components wherever practical. This is discussed in 3.5.1.1 and 3.5.1.2.

The potential missiles described in 3.5.1.2 were investigated and found not to prevent safe cold shutdown. Therefore, specific missile barriers were not required.

The pipe break/missile study evaluation currently underway at WNP-2 will consider some additional types of internall generated missiles. This expanded scope may require barriers for protection of essential systems and components. The expanded approach for internally generated missiles, criteria for required barriers, and systems requiring protection will be provided by amendment to the FSAR in the last quarter of 1981.

Q. 010.038
(3.5.1.2)

Section 3.5.1.1.2 of the FSAR states that missile trajectories are selected to encompass the most adverse conditions. It is not clear from the information provided in the FSAR what the trajectories of the credible primary missiles would be and what systems might be disabled by the missiles.

Provide the bases for selection of the probable missile trajectories and show the trajectories on the appropriate FSAR figure. Include a discussion on the system, component, or structure that could be damaged or disabled by a missile. The extent of damage from each missile should be discussed.

Response:

The pipe break/missile study evaluation currently under way for WNP-2 will provide the answer to NRC Question 010.038. FSAR Section 3.5 will be revised as appropriate in the last quarter of 1981.

Q. 010.039
(3.5.1.2)

Section 3.5.1.1.3.2 states that thermowells and sample probes do not present potential hazards as postulated missiles affecting safe shutdown.

Provide justification to support this position on the thermowells and sample probes.

Response:

As stated in the response to NRC Question 211.013, themowells and sample probes were investigated for their potential of becoming postulated missiles.

Thermowells in high energy systems were found to have connections holding the thermowell to the system that were conservative by many folds. Some thermowells and sample probes are discounted as potential missiles if the piping system pressures are low and there are large factors of safety against failure. The remaining thermowells and sample probes are still postulated as missiles. All equipment that could be contacted by the missile, assuming a 10° half angle cone for the target area, was assumed to fail.

Potential thermowell missiles were found not to prevent safe cold shutdown or the release of unacceptable amounts of radioactive materials assuming a single active component failure and loss of offsite power.

At WNP-2, sample probes are one-inch or less in diameter and meet the requirements of Regulatory Guide 1.11. Pipe breaks were not postulated in lines one-inch or less as recommended in Branch Technical Positions APCSB 3-1 and MEB 3-1.

Q. 010.040
(3.5.1)

The FSAR states that the water lines are "...tornado-hardened". State your criteria for protecting pipes located outside buildings from tornado missiles, including depth below grade requirements and provide drawings which show all pertinent tornado protection features as necessary.

Response:

As stated in Section 3.5.3, buried safety-related piping required for safe shutdown is ensured adequate protection from tornado-generated missiles. Analysis of potential damage is performed using "Tornado Design Considerations for Nuclear Power Plants" by Bates and Swanson, 1967 (Reference 3.5-8). A 5-foot embedment depth is calculated to be acceptable to ensure pipe integrity.

The standby service water piping exits the pumphouses at a centerline elevation of 435'3" and immediately turns down at a 45° angle to elevation 432 feet., where the piping is routed to the reactor building in high relative density Quality Class I backfill. Grade level is at 440'6", providing an embedment depth of over 7 feet from the top of the pipe. Where the pipe exits the pumphouses, a 1-1/2" asphaltic concrete road with a 6" base coarse and 2" leveling coarse bed provides additional protection from tornado-generated missiles. Additionally, the two standby service water loops are separated by at least 20 feet to preclude loss of redundancy. The standby service water pumphouses, shown in Figures 3.5-48 and 3.5-49, are protected from tornado-generated missiles. The standby service water piping and the tower makeup water system from the river are the only safety-related water piping systems outside of tornado-protected buildings. The tower make-up system is only required in the event that the spray ring headers in the ultimate heat sink are lost in the tornado. The tower make-up piping to the river also satisfies the five foot embedment criteria. Protection from tornadoes and tornado missiles in regard to such piping has also been previously addressed in response to Questions 010.024 and 010.027.

Though not technically a piping system in line with this question, the control room remote air intakes are, of course, located remote to tornado-hardened buildings. The intake structures themselves are tornado hardened, however, (see Figure 3.8-59) and the piping from the structures meets the five foot embedment criteria.

It should be noted that Chapters 3.5 and 3.6 will be revised in line with the on-going pipe break/missile study. The information in the response to this question will be more clearly incorporated into the FSAR text at that time.

Q. 010.041
(4.6)

Demonstrate that the scram discharge system meets the criteria enumerated in the Generic Safety Evaluation Report BWR Scram Discharge System, dated December 1, 1980.

Response:

The scram discharge system for WNP-2 has been evaluated against the Generic Safety Evaluation Report, "BWR Scram Discharge System", dated December 1, 1980. In short, the evaluation indicated that the WNP-2 scram discharge system needed upgrading in the following areas:

1) Addition of redundant vent and drain isolation valves;
2) Addition of redundant and diverse level instrumentation for scram;
3) Relocation and repiping of instrument piping directly to the scram instrument volume;
4) Addition of new surveillance and operating procedures.

A summary of our evaluation results is provided below:

FUNCTIONAL CRITERIA

1. The scram discharge volume (SDV) shall have sufficient capacity to receive and contain water exhausted by a full reactor scram without adversely affecting control-rod-drive scram performance.

   WNP-2 Compliance:

   WNP-2's SDV system is currently designed to meet the 3.34 gallons per drive requirement specified in the GE Design Specification 22A4260. This is an acceptable means of compliance as documented in the "Acceptable Compliance" statement for this item in the Generic SER.

SAFETY CRITERIA

1. No single active failure of a component or service function shall prevent a reactor scram, under the most degraded conditions that are operationally accepted.

   WNP-2 Compliance:

   The WNP-2 system has been designed to meet single failure criteria. The SDV is designed with an integral instrument volume (IV) which provides direct and immediate detection of liquid accumulation. The SDV instrumentation is redundant and single failure proof (including partial loss of service functions).

2. No single failure shall result in uncontrolled loss of reactor coolant.

   WNP-2 Compliance:

   A redundant air-operated vent valve and drain valve will be added on the SDV in series to insure system isolation during reactor scrams. This includes independent solenoid valves for each set of air-operated vent and drain valves.

3. The scram discharge system instrumentation shall be designed to provide redundancy, to operate reliably under all conditions, and shall not be adversely affected by hydrodynamic forces or flow characteristics.

   WNP-2 Compliance:

   Six additional diverse level sensors will be added to the SDV system to ensure diversity and redundancy in level monitoring and scram functions. Common cause failures will be considered in the selection of the instruments. This is in agreement with Alternative 3 of the "Acceptable Compliance" statement for this item in the Generic SER.

4. System operating conditions which are required for scram shall be continuously monitored.

   WNP-2 Compliance:

   The addition of the level switches described in 3 above and periodic surveillance testing of the instruments will provide a continuous means of monitoring the SDV liquid level and insuring instrument reliability. This is an acceptable means of compliance as documented in the "Acceptable Compliance" statement for this item in the Generic SER.

5. Repair, replacement, adjustment, or surveillance of any system component shall not require the scram function to be bypassed.

   WNP-2 Compliance:

   During routine surveillance testing, instrument repair or calibration the associated logic will be placed in a half-scram (1 out of 2) configuration in accordance with the plants technical specifications. This is an acceptable means of compliance as documented in the "Acceptable Compliance" statement for this item in the Generic SER.

OPERATIONAL CRITERIA

1. Level instrumentation shall be designed to be maintained, tested, or calibration during plant operation without causing a scram;

2. The system shall include sufficient supervisory instrumentation and alarms to permit surveillance of system operation;

3. The system shall be designed to minimize the exposure of operating personnel to radiation;

4. Vent paths shall be provided to assure adequate draining in preparation for scram reset;

5. Vent and drain functions shall not be adversely affected by other system interfaces. The objective of this requirement is to preclude water backup in the scram instrument volume which could cause spurious scram.

WNP-2 Compliance:

1. The system logic is designed as a one out of two, twice configuration. Each of the associated instrument channels is capable of being separately isolated for maintenance, testing or calibration without inadvertently scramming the reactor.

2. The SDV is provided with a high liquid level alarm on each IV to alert the operator to liquid accumulation in the SDV.

3. The SDV system has been designed in accordance with GE design specification 22A4260 to minimize the exposure of operating personnel to radiation. In addition, the system is being reviewed as part of the WNP-2 ALARA program.

4. The SDV vents directly to the reactor building atmosphere and is independent from other plant vent system.

5. The vent and drain system for the SDV is totally independent from other plant systems, and is therefore not susceptible to blockage or water buildup through system interfaces.

DESIGN CRITERIA

1. The scram discharge headers shall be sized in accordance with GE OER-54 and shall be hydraulically coupled to the instrumented volume(s) in a manner to permit operability of the scram level instrumentation prior to loss of system function. Each system shall be analyzed based on a plant-specific maximum inleakage to ensure that the system function is not lost prior to initiation of automatic scram. Maximum inleakage is the maximum flow rate through the scram discharge line without control-rod motion summed over all control rods. The analysis should show no need for vents or drains.

   WNP-2 Compliance:

   WNP-2's IVs have been designed as vertical extensions attached directly to the SDV. This configuration provides a direct hydraulic couple between the SDV and IVs and insures immediate and continuous liquid level monitor in the SDV. This is an acceptable means of compliance as documented in the "Acceptable Compliance" statement for this item in the Generic SER.

2. Level instrumentation shall be provided for automatic scram initiation while sufficient volume exists in the scram discharge volume.

   WNP-2 Compliance:

   WNP-2's SDV is adequately coupled to the IV to allow proper instrument operation. The SDV instrument setpoint for scram was established to insure an available volume of 3.34 gallons per drive (185 drives). This is an acceptable means of compliance as documented in the "Acceptable Compliance" statement for this item in the Generic SER.

3. Instrumentation taps shall be provided on the vertical instrument volume and not on the connected piping.

   WNP-2 Compliance:

   All the WNP-2 SDV instrumentation will be relocated and repiped directly to the IV instead of the vent and drain piping. Procedures will be modified to include functional testing of SDV level instrumentation after each scram. This is an acceptable means of compliance as documented in the "Acceptable Compliance" statement for this item in the Generic SER.

4. The scram instrumentation shall be capable of detecting water accumulation in the instrumented volume(s) assuming a single active failure in the instrumentation system or the plugging of an instrument line.

WNP-2 Compliance:

The addition of the redundant and diverse instruments described under Safety Criterion 3 and rerouting of the instrument piping to the IV provide an acceptable means of compliance as documented in the "Acceptable Compliance" statement for this item in the Generic SER.

5. Structural and component design shall consider loads and conditions including those due to fluid dynamics, thermal expansion, internal pressure, seismic considerations, and adverse environments.

WNP-2 Compliance:

The WNP-2 SDV design compliance with the latest GE design criteria as outlined in GE Design Specification 22A4260. In addition, the system will be reviewed as part of the equipment qualification program.

6. The power-operated vent and drain valves shall close under loss of air and/or electric power. Valve position indication shall be provided in the control room.

WNP-2 Compliance:

WNP-2's present design configuration meets these requirements.

7. Any reductions in the system piping flow path shall be analyzed to assure system reliability and operability under all modes of operation.

WNP-2 Compliance:

WNP-2's SDV header system is designed as a continually expanding path from the 185 3/4" individual scram discharge (withdrawal) lines to one of two integrated SDV/IV systems (one system per approximately half the drives). Each integrated SDV/IV system consists of a continuously downsloping piping run expanding from the SDV (consisting of seven 6" return headers from the individual hydraulic control unit (HCU) banks to an 8" combined return header) to the 12" vertically oriented IV. The location where blockage need be assumed (piping less than 2" diameter) is in the 3/4" discharge line from the individual HCU. Blockage here would only cause failure of one control rod to insert. This is an acceptable consequence for a single failure and has been evaluated as part of the plant design basis. Accordingly, this design complies with the "Acceptable Compliance" statement for this item in the Generic SER.

8. System piping geometry (i.e., pitch, line size, orientation) shall be such that the system drains continuously during normal plant operation.

   WNP-2 Compliance:

   The WNP-2 SDV has been designed to insure a positive downward slope of scram header and drain piping.

9. Instrumentation shall be provided to aid the operator in the detection of water accumulation in the instrumented volume(s) prior to scram initiation.

   WNP-2 Compliance:

   Each IV is provided with high liquid level and rod block instrumentation attached directly to it. The generic SER states that this is acceptable.

10. Vent and drain line valves shall be provided to contain the scram discharge water, with a single active failure and to minimize operational exposure.

    WNP-2 Compliance:

    As stated under Safety Criterion 2, redundant air-operated vent and drain valves will be provided for system isolation. This is an acceptable means of compliance as documented in the "Acceptable Compliance" statement for this item in the Generic SER.

SURVEILLANCE CRITERIA

1. Vent and drain valves shall be periodically tested.

WNP-2 Compliance:

The vent and drain valves will be tested in accordance with the plant technical specification to verify valve closure in less than 30 seconds (current GE specification).[1] This is an acceptable means of compliance as documented in the "Acceptable Compliance" statement for this item in the Generic SER.

2. Verifying and level detection instrumentation shall be periodically tested in place.

WNP-2 Compliance:

The SDV instrumentation will be tested in accordance with the plants technical specification which will include post scram testing to verify instrument operability.[1]

3. The operability of the entire system as an integrated whole shall be demonstrated periodically and during each operating cycle, by demonstrating scram instrument response and valve function at pressure and temperature at approximately 50% control-rod density.

WNP-2 Compliance:

Surveillance testing will be performed in accordance with the plants technical specifications.

[1] The plant technical specifications will be based on the Standard Technical Specifications for Boiling Water Reactors, NUREG-0123, provided by the NRC.

Q. 010.042
(4.6)
RSP

Demonstrate that a slow or partial loss of air pressure to the scram discharge valves will not result in the following:

1) Rapid filling of both the scram discharge volume and the instrument volume due to the lifting of most or all scram discharge valves, with consequent loss of adequate scram discharge volume.

2) Loss of reactor coolant due to the combination of lifting of most or all scram discharge valves, without compensating closure of the vent and drain valves, with consequent environmental effects inside containment.

Unless it can be demonstrated that no adverse effects can result, a system shall be provided and describeo in this section to protect against these two conditions.

Response:

1) The WNP-2 scram discharge instrument volume (IV) is adequately hydraulically coupled to the scram discharge volume (SDV), i.e., the IV is connected directly to the SDV with piping of a diameter equal to or greater than the diameter of the SDV headers. This allows for direct and immediate detection of liquid buildup so that the ability to scram is ensured, even in the event of lifting of most or all of the scram valves, when the water buildup reaches scram initiation level in the IV.

   Note: The basis of the instrument volume high level scram setpoint and the IV/SDV physical arrangement provides for scram action before significant scram discharge volume reduction occurs which could affect scram capability.

2) The partial loss of air pressure does not result in the uncontrolled release of reactor coolant to the reactor building should all or most of the scram discharge valves lift. When the water buildup reaches scram initiation level in the IV, a scram signal is produced.

This will cause the air supply to the vent and drain valves to vent, thereby ensuring that the vent and drain valves close and isolate. For leakage rates which do not result in buildup in the IV, the leak will drain to the reactor building equipment drain system. The drain system will alarm for leakage rates greater than five (5) gallons per minute. The operator can then take appropriate action, e.g., isolate the leak, scram the reactor, increase air pressure, etc., as required.

See response to Question 010.042 for additional information.

Q. 010.044
(4.6.1.1.2.4)

Table 1.3-8 indicates specific design changes from the PSAR to the FSAR for the CRD system. The design changes for the CRD return line modification addressed in Question 211.019, have not been included in the text description of the FSAR and Figures 4.6-5a, 4.6-5b, and 4.6-6a have not been revised. Revise the text description in the FSAR to reflect the specific design changes in Table 1.3-8 for the CRD system and modify the above figures accordingly.

Response:

Figures 4.6-5a, 4.6-5b, 4.6-6a, 4.6-6b, and 4.6-6c are revised to reflect the CRD return line modification. Figures 4.6-6d and 4.6-6e are deleted. Also, text sections 4.6.1.1.2.4, 4.6.1.1.2.4.1, 4.6.1.1.2.4.2.3, 4.6.1.1.2.4.2.4 and 4.6.2.3.2.2.8 are revised accordingly. Text section 4.6.1.1.2.4.2.5 is deleted.*

*FSAR revised page changes attached.

FSAR REVISIONS

e. Metal piston rings are Haynes 25 alloy.

f. Certain wear surfaces are hard-faced with Colmonoy 6.

g. Nitriding by a proprietary new Malcomizing process and chromium plating are used in certain areas where resistance to abrasion is necessary.

h. The drive piston head is made of Armco 17-4Ph.

Pressure-containing portions of the drives are designed and fabricated in accordance with requirements of Section III of the ASME Boiler and Pressure Vessel Code.

4.6.1.1.2.4 Control Rod Drive Hydraulic System

The control rod drive hydraulic system (Figures 4.6-5a, b) supplies and controls the pressure and flow to and from the drives through hydraulic control units (HCU). The water discharged from the drives during a scram flows through the HCUs to the scram discharge volume. The water discharged from a drive during a normal control rod positioning operation flows through the HCU, the exhaust header, and ~~into the cooling water header.~~ is returned to ~~the~~ reactor vessel via the HCUs of non-moving drives. There are as many HCUs as the number of control rod drives.

4.6.1.1.2.4.1 Hydraulic Requirements

The CRD hydraulic system design is shown in Figures 4.6-5a, b, and 4.6-6. The hydraulic requirements, identified by the function they perform, are as follows:

a. An accumulator hydraulic charging pressure of approximately 1400 to 1500 psig is required. Flow to the accumulators is required only during scram reset or system startup.

b. Drive water header pressure of approximately 260 psi above reactor vessel pressure is required. A flow rate of approximately 4 gpm to insert a control rod and 2 gpm to withdraw a control rod is required.

c. Cooling water to the drives is required at approximately 20 psi above reactor vessel pressure and at a flow rate of approximately ~~0.20 to~~ 0.34 gpm per drive unit. (Cooling water to a drive can be interrupted for short periods without damaging the drive.)

~~d. The drive header pressure will be no more than 5 psi above the cooling water header pressure. (The pressure in this line must be kept as low as possible to avoid interference with normal drive movement.)~~

d ~~e.~~ The scram discharge volume is sized to receive and contain all the water discharged by the drives during a scram; a minimum volume of 3.34 gal. per drive is required (excluding the instrument volume).

4.6.1.1.2.4.2 System Description

The CRD hydraulic systems provide the required functions with the pumps, filter, valves, instrumentation, and piping shown in Figures 4.6-5a, b and described in the following paragraphs.

Duplicate components are included, where necessary, to assure continuous system operation if an in-service component requires maintenance.

4.6.1.1.2.4.2.1 Supply Pump

One supply pump pressurizes the system with water from a condensate supply header which takes suction from the condensate treatment system and/or condensate storage tanks depending on plant operation. One spare pump is provided for standby. A discharge check valve prevents backflow through the nonoperating pump. A portion of the pump discharge flow is diverted through a minimum flow bypass line to the condensate storage tank. This flow is controlled by an orifice and is sufficient to prevent immediate pump damage if the pump discharge is inadvertently closed.

Condensate water is processed by two filters in the system. The pump suction filter is a disposable element type with a 25-micron absolute rating. A 250-micron strainer in the filter bypass line protects the pump when the filter is being serviced. The drive water filter downstream of the pump is a cleanable element type with a 50-micron absolute rating. A differential pressure indicator and control room alarm monitor the filter element as it collects foreign materials.

4.6.1.1.2.4.2.2 Accumulator Charging Pressure

Accumulator charging pressure is established by the discharge pressure of the system supply pump. During scram the scram inlet (and outlet) valves open and permit the stored energy in the accumulators to discharge into the drives. The resulting pressure decrease in the charging water header allows the CRD supply pump to "run out" (i.e., flow rate to increase substantially) into the control rod drives via the charging water header. The flow sensing system upstream of the accumulator charging header detects high flow and closes the flow control valve. This action maintains increased flow through the charging water header.

Pressure in the charging header is monitored in the control room with a pressure indicator and high pressure alarm.

During normal operation the flow control valve maintains a constant system flow rate. This flow is used for drive flow, drive cooling, and system stability.

4.6.1.1.2.4.2.3 Drive Water Pressure

Drive water pressure required in the drive header is maintained by the ~~drive~~ drive/cooling water pressure control valve, which is manually adjusted from the control room. A flow rate of approximately 6 gpm (the sum of the flow rate required to insert and withdraw a control rod) normally passes from the drive water pressure stage through two solenoid operated stabilizing valves (arranged in parallel) and then goes into the cooling water header. The flow through one stabilizing valve equals the drive insert flow; that of the other stabilizing valve equals the drive withdrawal flow. When operating a drive, the required flow is diverted to that drive by while closing the appropriate stabilizing valve. Thus, flow through the ~~drive~~ drive/cooling water pressure control valve is always constant.

Sa[illegible] at the s[illegible] time, ope[illegible] the ~~[illegible]~~ drive direc[illegible] control an[illegible] exhaust solenoid valves.

Flow indicators in the drive water header and in the line downstream from the stabilizing valves allow the flow rate through the stabilizing valves to be adjusted when necessary. Differential pressure between the reactor vessel and the drive pressure stage is indicated in the control room.

711.130-4

Add Insert 'A'

4.6.1.1.2.4.2.4 Cooling Water Header

~~The cooling water header is located just upstream of the return line control valve. An automatic pressure regulating valve controls the pressure in this header, which is set to produce the desired cooling water flow to the drives. A flow indicator in the control room monitors cooling water flow. A differential pressure indicator in the control room indicates the difference between reactor vessel pressure and drive cooling water pressure. Although the drives can function without cooling water, seal life is shortened by long term exposure to reactor temperatures. The temperature of each drive is recorded in the control room, and excessive temperatures are annunciated.~~

~~4.6.1.1.2.4.2.5 Return Line~~

~~The return line routes excess flow from the control rod drive hydraulic system (water not used for charging of accumulators, movement of drives or cooling) through the reactor water cleanup system to the reactor feedwater line. A pressure control valve in this line is manually adjusted from the control room to produce the desired pressure. The flow through this valve is virtually constant. Therefore, once adjusted, the drive pressure control valve and the return water control valve can maintain their required pressure independent of reactor pressure.~~

renumber as req'd

4.6.1.1.2.4.2.6 Scram Discharge Volume

The scram discharge volume consists of header piping which connects to each HCU and drains into an instrument volume. The header piping is sized to receive and contain all the water discharged by the drives during a scram, independent of the instrument volume.

During normal plant operation the scram discharge volume is empty, and vented to atmosphere through its open vent and drain valve. When a scram occurs, upon a signal from the safety circuit these vent and drain valves are closed to conserve reactor water. Lights in the control room indicate the position of these valves.

INSERT. A

The cooling water header is located downstream from the drive/cooling water pressure valve. The drive/cooling water pressure control valve is manually adjusted from the control room to produce the required drive/cooling water pressure balance.

The flow through the flow control valve is virtually constant. Therefore, once adjusted, the drive/cooling water pressure control valve will maintain the correct drive pressure and cooling water pressure, independent of reactor vessel pressure. Changes in setting of the pressure control valves are required only to adjust for changes in the cooling requirements of the drives, as the drive seal characteristics change with time. A flow indicator in the control room monitors cooling water flow. A differential pressure indicator in the control room indicates the difference between reactor vessel pressure and drive cooling water pressure. Although the drives can function without cooling water, seal life is shortened by long term exposure to reactor temperatures. The temperature of each drive is indicated and recorded, and excessive temperatures are annunciated in the control room.

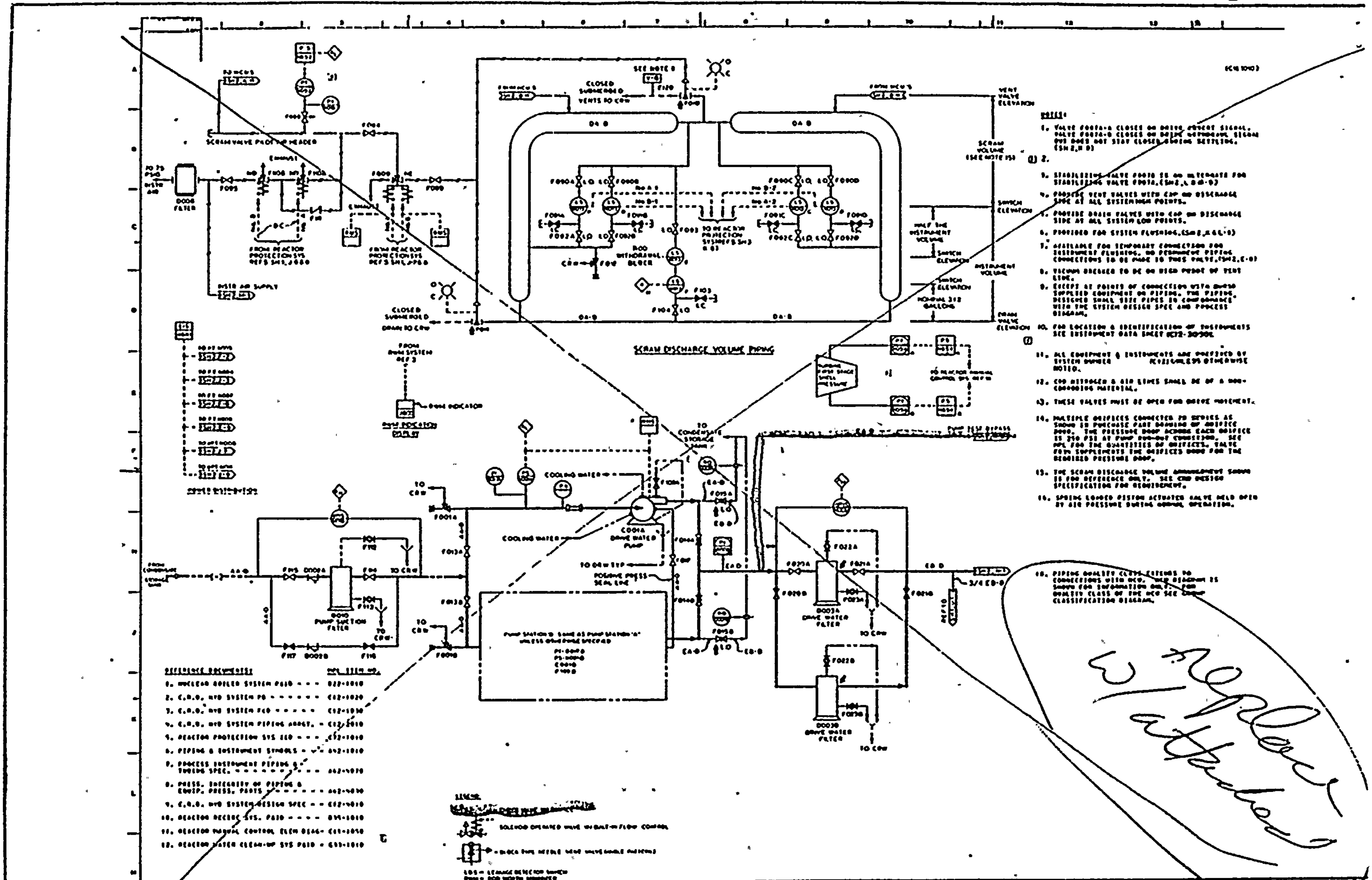
If the plug were to blow out while the drive was latched, there would be no control rod motion. No pressure differential would exist across the collet piston to unlatch the collet. As in the previous failure, reactor water would flow past the velocity limiter, down the annulus between the drive and thermal sleeve, through the vessel ports and drilled passage, through the ball check valve cage and out the open plug hole to the drywell. The leakage calculations indicate the flow rate would be 350 gpm. This calculation assumes liquid flow, but flashing of the hot reactor water to steam would reduce this rate to a lower value. Drive temperature would rapidly increase and initiate an alarm in the control room.

If the plug failure were to occur during control rod withdrawal, (it would not be possible to unlatch the drive after such a failure) the collet would relatch at the first locking groove. If the collet were to stick, calculations indicate the control rod withdrawal speed would be 11.8 feet per second. There would be a large retarding force exerted by the velocity limiter due to a 35 psi pressure differential across the velocity limiter piston.

4.6.2.3.2.2.8 ~~Drive~~ Drive/Cooling Water Pressure Control Valve Closure (Reactor Pressure, 0 psig)

The pressure to move a drive is generated by the pressure drop of practically the full system flow through the drive/cooling water pressure control valve. This valve is either a motor operated valve or a standby manual valve; either one is adjusted to a fixed opening. The normal pressure drop across this valve develops a pressure 260 psi in excess of reactor pressure.

If the flow through the ~~drive~~ drive/cooling water pressure control valve were to be stopped, as by a valve closure or flow blockage, the drive pressure would increase to the shutoff pressure of the supply pump. The occurrence of this conditon during withdrawal of a drive at zero vessel pressure will result in a drive pressure increase from 260 psig to no more than 1750 psig. Calculations indicate that the drive would accelerate from 3 in./sec to approximately 6.5 in./sec. A pressure differential of 1670 psi across the collet piston would hold the collet unlatched. Flow would be upward, past the velocity limiter piston, but retarding force would be negligible. Rod movement would stop as soon as the driving signal was removed.



WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

CONTROL ROD DRIVE HYDRAULIC SYSTEM P&ID

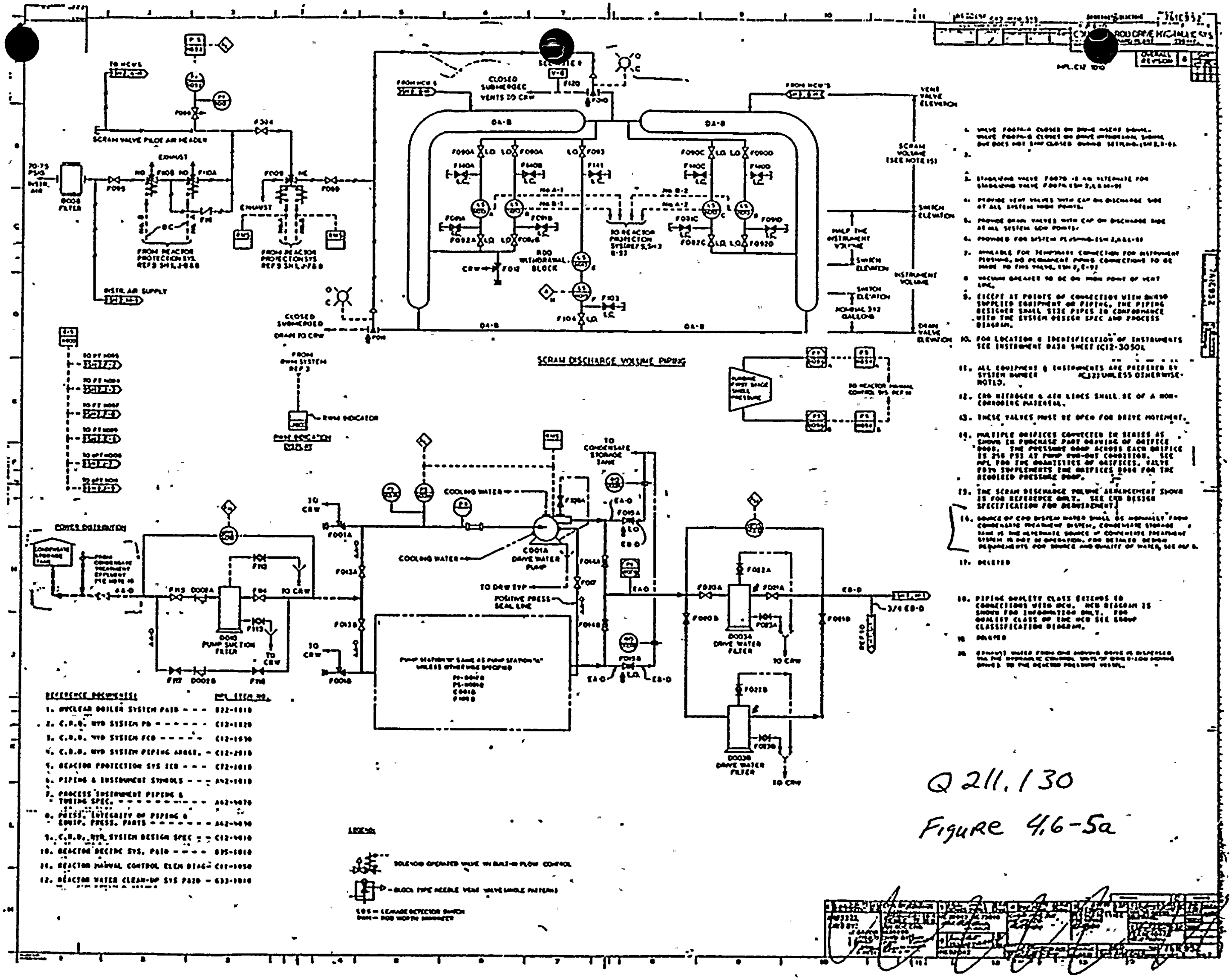


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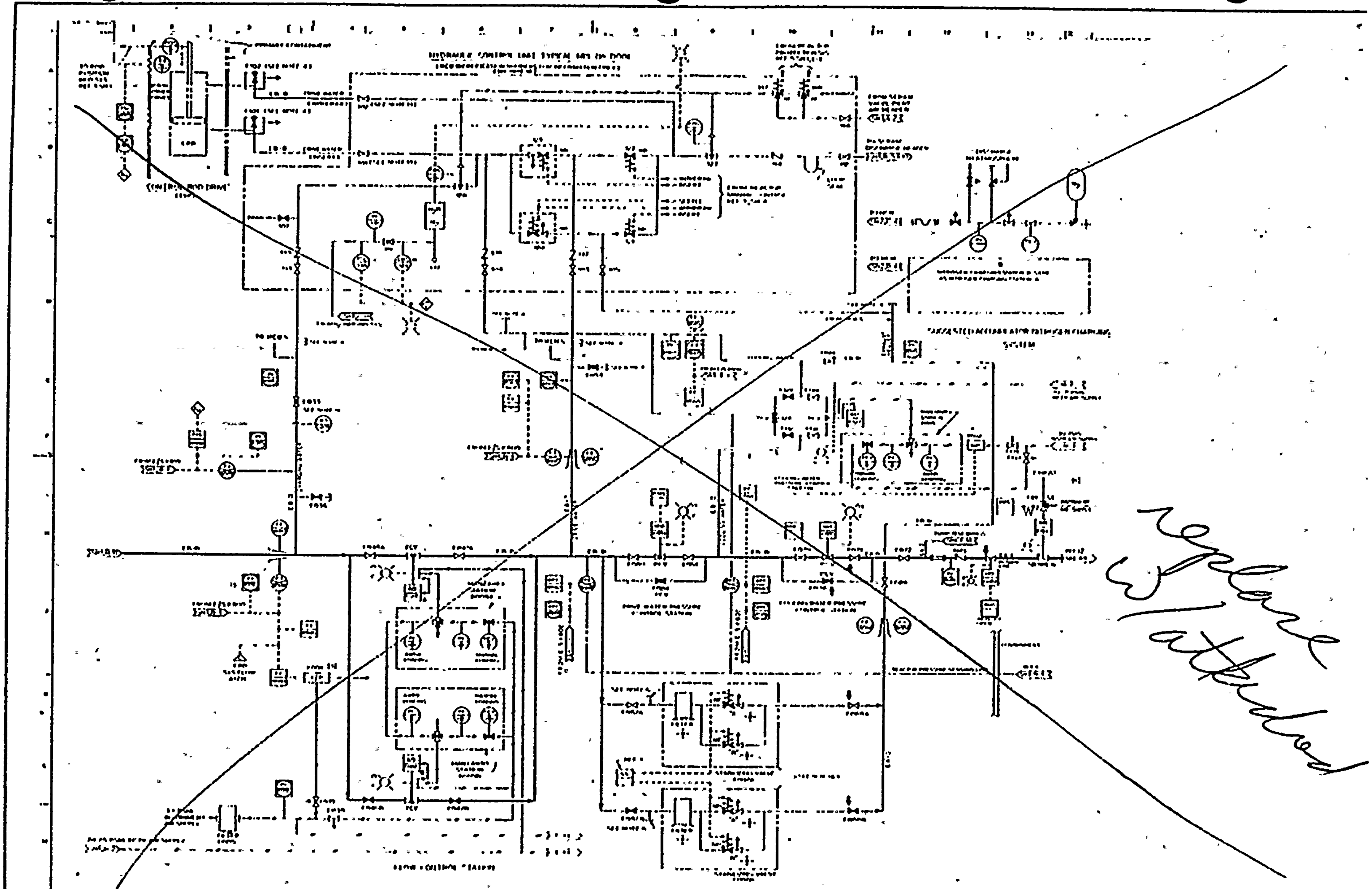
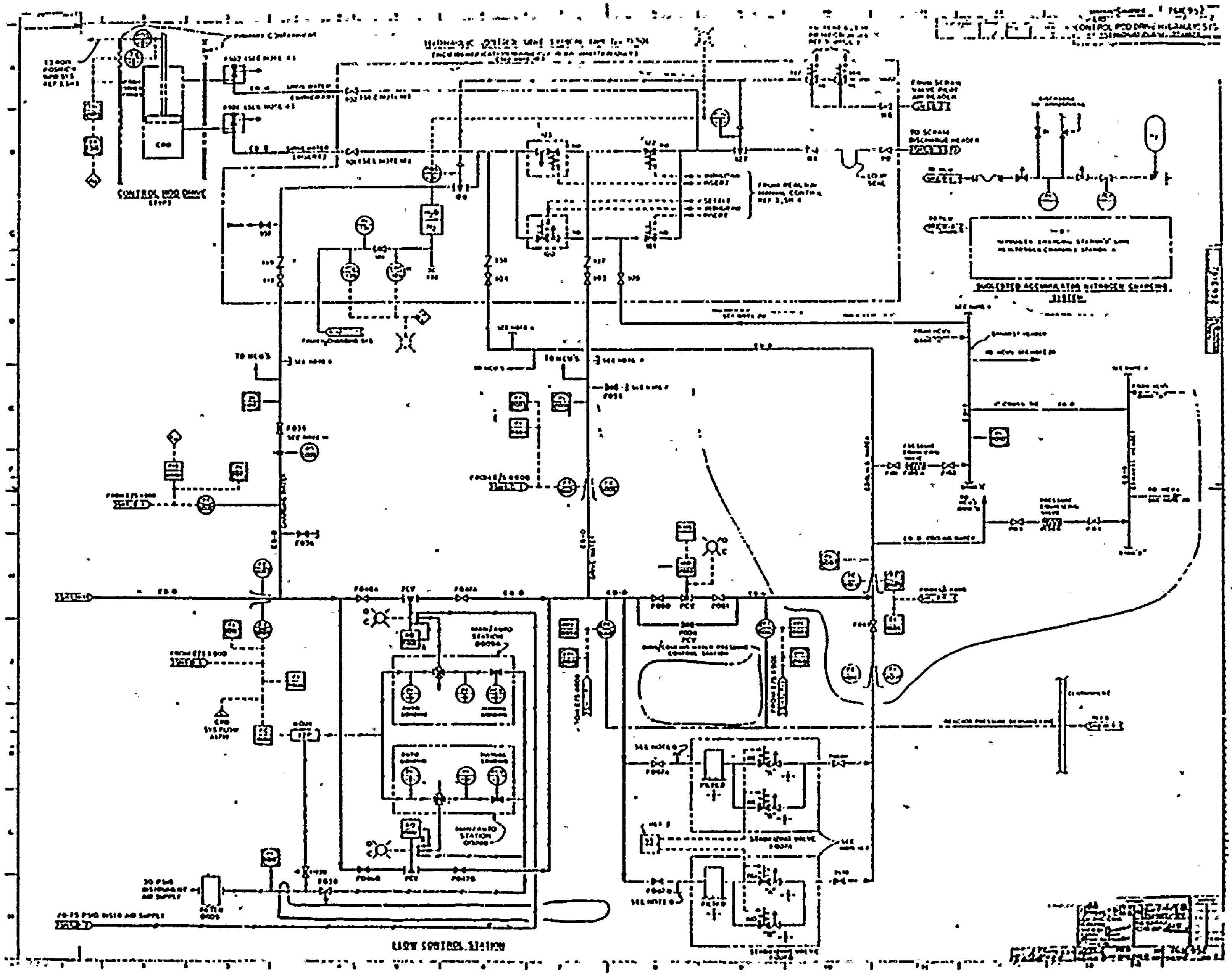
Figure 4.6-5a

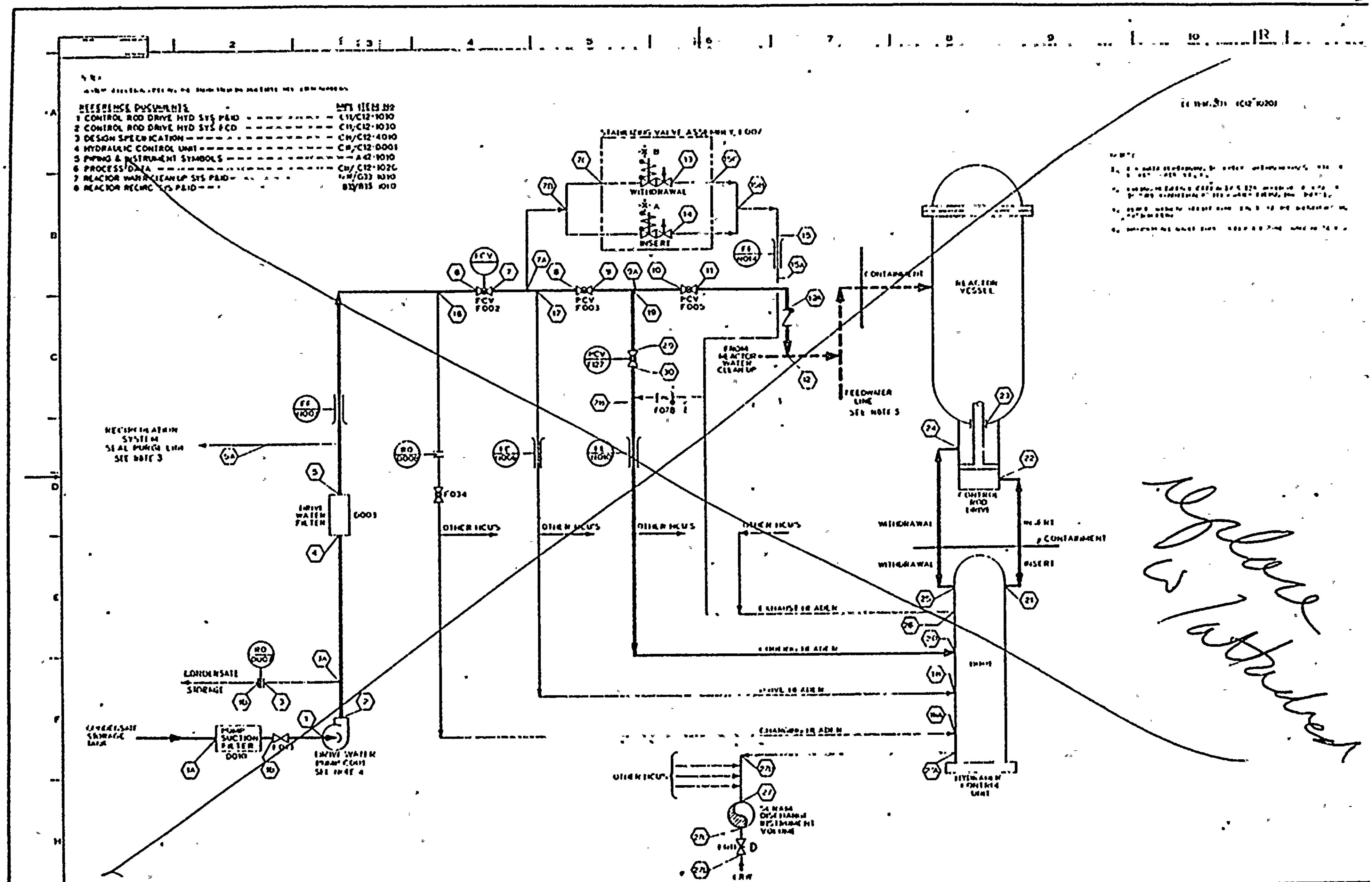


Replace w/attached

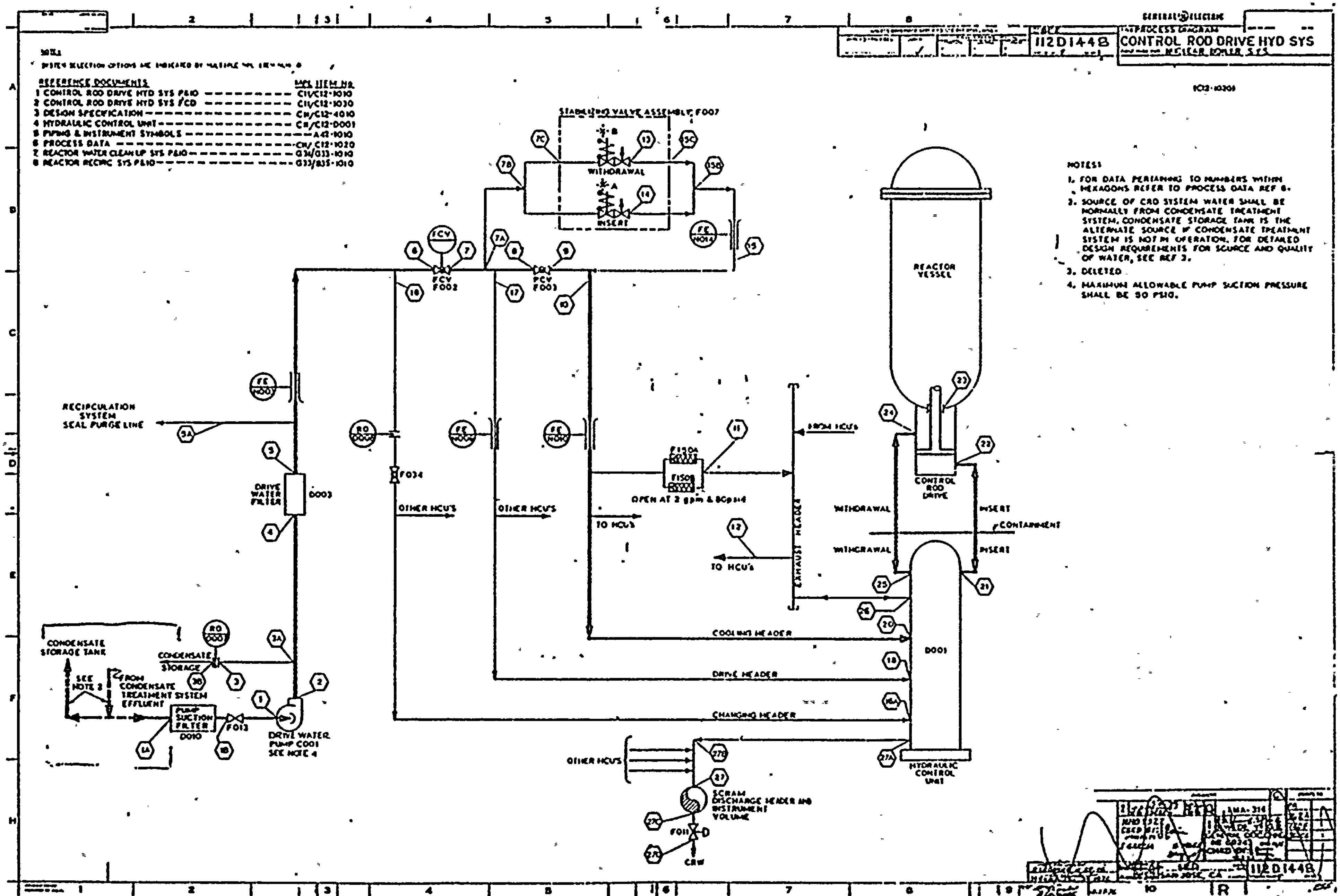
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL ROD DRIVE HYDRAULIC SYSTEM P&ID



Replaces FSAR
Figure 4.6-5b



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL ROD DRIVE SYSTEM PROCESS DIAGRAM



Replaces FSAR Figure 4.6-6a

WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

MODE A NORMAL OPERATION

LOCATION	1A	1	2	3	4	5	5A	6	7	8	9
FLOW, GPM	106.9	106.9	106.9	19.0	87.9	87.9	10.0	77.9	77.9	71.9	71.9
PRESS. PSIG	21.	19.	1429.	1429.	1401.	1388.	1388.	1383.	PR+260.	PR+260.	PR+60.

LOCATION	10	11	12	13	14	15	16	17	18	19	20
FLOW, GPM	15.0	15.0	15.0	2.0	4.0	6.0	0.	/	/	56.9	.34MAX
PRESS. PSIG	PR+60.	PR+50.	PR+50.MAX	PR+20.MAX	PR+20.MAX	PR+20.MAX	1365.			PR+60.	PR+15.

LOCATION	21	22	23	24	25	26	27	28	29	30	
FLOW, GPM	.34MAX	.34MAX	.34MAX	0	/	0	0	62.9	56.9	56.9	
PRESS. PSIG	PR+14.	PR+14.	PR	PR		PR+20.MAX	0	PR+20.MAX	PR+60.	PR+20.MAX	

CONDITIONS
1. DRIVES LATCHED.
2. PRESSURE OF REACTOR (PR) AT 1000 PSIG.
3. MAXIMUM COOLING FLOW TO DRIVES.

MODE A SIZES THE COOLING WATER HEADERS
MINIMUM REQUIRED PRESSURE AT POSITION 1A IS SHOWN.
PRESSURE AT LOCATION 16 SHALL NOT EXCEED 1510 PSIG
PRESSURE AT LOCATION 28 SHALL NOT EXCEED PR+20.
LINE LOSS FROM LOCATION 30 TO LOCATION 20 SHALL NOT EXCEED 3 PSIG.

MODE B ROD INSERTION

LOCATION	1A	1	2	3	4	5	5A	6	7	8	9
FLOW, GPM	"	"	"	"	"	"	"	"	"	"	"
PRESS. PSIG	"	"	"	"	"	"	"	"	"	"	"

LOCATION	10	11	12	13	14	15	16	17	18	19	20
FLOW, GPM	"	"	15	"	0	2.0	"	4.0	4.0	"	/
PRESS. PSIG	"	"	"	"	"	"	"	PR+260.	PR+250.	"	"

LOCATION	21	22	23	24	25	26	27	28	29	30	
FLOW, GPM	4.0	4.0	1.3	.7	.7	.7	"	56.9	"	"	
PRESS. PSIG	PR+81.	PR+90.	"	PR+20.MAX	PR+20.MAX	PR+20.MAX	"	PR+20.MAX	"	"	

CONDITIONS:
1. DRIVES INSERTING.
2. PRESSURE OF REACTOR (PR) AT 1000 PSIG.
3. MAXIMUM DRIVING FLOW TO DRIVES

MODE B SIZES THE DRIVE WATER HEADERS

FOR REVISIONS SEE SHEET 1

THIS SHEET TO BE USED WITH PROCESS DIAGRAM 112D1448

CONTROL ROD DRIVE SYSTEM

FIGURE 4.6-6b

GENERAL ELECTRIC

PROCESS DIAG DATA

CONTROL ROD DRIVE HYD SYSTEM

112D1448AD

MPL NO. C12-1020

OVERALL REVISION 2

MODE A NORMAL OPERATION

LOCATION	1A	1	2	3	4	5	5A	6	7	8	9	10	11	12	13
FLOW, GPM	93	93	93	20	73	73	10	63	63	57	57	63	0	0	2
PRESSURE PSIG	21	19	1487	1487	1476	1462	1462	1455	PR + 260	PR + 260	PR + 30	PR + 30	PR	PR	PR + 30

LOCATION	14	15	16	17	18		20	21	22	23	24	25	26	27	
FLOW, GPM	4	6	0	0	0		.34 MAX	.34 MAX	.34 MAX	.34 MAX	0	0	0	0	
PRESSURE PSIG	PR + 30	PR + 30	1455				PR + 15	PR + 14	PR + 14	PR	PR		PR	0	

CONDITIONS:

1. DRIVES LATCHED
2. PRESSURE OF REACTOR (PR) AT 1000 PSIG.
3. MAXIMUM COOLING FLOW TO DRIVES. MINIMUM REQUIRED PRESSURE AT POSITION 1A IS 20 FEET OF WATER AT 200 GPM.

MODE A SIZES THE COOLING WATER HEADERS.

LINE LOSS FROM LOCATION 10 TO LOCATION 20 SHALL NOT EXCEED 3 PSIG.

(FOR NOTES SEE SHEET 2.)

MODE B ROD INSERTION

LOCATION	1A	1	2	3	4	5	5A	6	7	8	9	10	11	12	13
FLOW, GPM	93	93	93	20	73	73	10	63	63	57	57	59	0	.7	2
PRESSURE PSIG	21	19	1487	1487	1476	1462	1462	1455	PR + 260	PR + 260	PR + 30	PR + 30	PR + 8	PR + 8	PR + 30

LOCATION	14	15	16	17	18		20	21	22	23	24	25	26	27	
FLOW, GPM	0	2	0	4	4		0	4	4	1.3	.7	.7	.7	0	
PRESSURE PSIG	PR + 30	PR + 30	1455	PR + 260	PR + 250		PR + 15	PR + 31	PR + 30	PR	PR + 20 MAX	PR + 20 MAX	PR + 8 MAX	0	

CONDITIONS:

1. DRIVE INSERTING
2. PRESSURE OF REACTOR (PR) AT 1000 PSIG.
3. MAXIMUM DRIVING FLOW TO DRIVES

MODE B SIZES THE DRIVE WATER HEADERS.

MODE C SCRAM

LOCATION	1A	1	2	3	4	5	5A	6	7	8	9	10	11	12	13
FLOW, GPM	45	45	45	20	25	25	10	15	15	15	15	15	15	14.3	0
PRESSURE PSIG	21	21	1550	1550									SEE NOTE 3	"SEE" NOTE 3	

LOCATION	14	15	16	17	18		20	21	22	23	24	25	26	27	
FLOW, GPM	0	0	0	0	0		0	30	30	-3.6	30	30	0.1 + SEE NOTE 3	APPROX 55CS	
PRESSURE PSIG								1167 MIN	731 MIN	PR	256 MAX	34		65 MAX SEE NOTE 10	

CONDITIONS:

1. DRIVES SCRAMMING
2. PRESSURE OF REACTOR (PR) AT 1000 PSIG.
3. FLOWS BASED ON MAXIMUM ROD VELOCITY OF 85 INCHES PER SECOND.

MODE C SIZES THE INSERT AND WITHDRAW LINES.

MODE D SCRAM COMPLETED

LOCATION	1A	1	2	3	4	5	5A	6	7	8	9	10	11	12	13
FLOW, GPM	200	200	200	20	180	180	10	15	15	15	15	15	16	14.9	0
PRESSURE PSIG	21	19	1210						>PR	>PR	>PR	>PR	>PR	>PR	

LOCATION	14	15	16	17	18		20	21	22	23	24	25	26	27	
FLOW, GPM	0	0	155	0	0		0	0.92	0.92	0.92	SEE NOTE 5	SEE NOTE 5	0.1	.0	
PRESSURE PSIG			348					76	76	PR	65 MAX	65 MAX		65 MAX	

SEE NOTE 10

CONDITIONS:

1. SCRAMMING OF DRIVES COMPLETED
2. PRESSURE OF REACTOR (PR) AT 0 PSIG.
3. MAXIMUM CRD SUPPLY PUMP FLOW.

MODE D SIZES THE PUMP SUCTION LINE.

NOTE: MINIMUM ACCUMULATOR PRECHARGE PRESSURE IS 565 PSIG.

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I GARCIA

112D1448AD

Replaces FSAR Figure 4.6-6b

replace w/attached

MODE C SCRAM

LOCATION	1A	1	2	3	4	5	5A	6	7	8	9
FLOW, GPM	"	"	"	"	"	"	"	"	"	"	"
PRESS. PSIG	"	"	"	"	"	"	"	"	"	"	"

LOCATION	10	11	12	13	14	15	16	17	18	19	20
FLOW, GPM	"	"	"	"	"	"	"	"	"	/	0
PRESS. PSIG	"	"	"	"	"	"	"	"	"		"

LOCATION	21	22	23	24	25	26	27	28	29	30	
FLOW, GPM	90.	90.	-3.6	29.6	29.6	"	APPROX.5476	"	"	"	
PRESS. PSIG	1167.MIN	731.MIN	"	256. MAX	94.	"	7.7				

CONDITIONS:

1. DRIVE SCRAMMING
2. PRESSURE OF REACTOR (PR) AT 1000. PSIG
3. FLOWS BASED ON MAXIMUM ROD VELOCITY OF 85 INCHES PER SECOND.

MODE C SIZES THE INSERT AND WITHDRAW LINES

MODE D SCRAM COMPLETED

LOCATION	1A	1	2	3	4	5	5A	6	7	8	9
FLOW, GPM	200.	200.	200.	18.	181.	181.	10.	5.	5.	5.	5.
PRESS. PSIG	6.	-4.	1141.	1141.	1064.	1010.	1010	988.	PR	PR	PR

LOCATION	10	11	12	13	14	15	16	17	18	19	20
FLOW, GPM	5.	5.	5.	/	/	/	166.	/	/	/	/
PRESS. PSIG	PR	PR	PR				988.				

LOCATION	21	22	23	24	25	26	27	28	29	30	
FLOW, GPM	0.95	0.95	1.48	-0.53	-0.53	/	0	/	/	/	
PRESS. PSIG	76.	76.	"	65.MAX	65.MAX		65.MAX				

CONDITIONS:

1. SCRAMMING OF DRIVES COMPLETED.
2. PRESSURE OF REACTOR (PR) AT 0 PSIG.
3. MAXIMUM CRD SUPPLY PUMP FLOW.

MODE D SIZES THE PUMP SUCTION LINE

MINIMUM ACCUMULATOR PRECHARGE PRESSURE IS 565. PSIG.

SEE SHEET 1 FOR REVISIONS

WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

CONTROL ROD DRIVE SYSTEM

FIGURE 4.6-6c

GENERAL ELECTRIC

PROCESS DIAG DATA

CONTROL ROD DRIVE HYD SYSTEM

112D1448AD

TABLE 1

LOCATION	1A-1B	1B-1	2-6	3A-3B	6-9	7A-7B	7B-7C
DESIGN PRESS. (PSIG.)	150	150	1750	1750	1750	1750	1750
DESIGN TEMP. (DEG F)	150	150	150	150	150	150	150
ESTIMATED LINE SIZE (INCHES)	4	4	2	1	1.5	1	0.75
LOCATION	10-20	11-12	13B-13C	15-15B	16-16A	17-18	12-26
DESIGN PRESS. (PSIG.)	1750	1750	1750	1750	1750	1750	1750
DESIGN TEMP. (DEG F)	150	150.	150	150	150	150	150
ESTIMATED LINE SIZE (INCHES)	2**	1	0.75	1	2	1	1
LOCATION	21-22	24-25	27A-27B	27B-27	27-27C	27C-27D	5A
DESIGN PRESS. (PSIG.)	1750	1750	1250	1250	1250	1250	1750
DESIGN TEMP. (DEG F)	150	550	280	280	280	280	150
ESTIMATED LINE SIZE (INCHES)	1	0.75	0.75	*	10	2	.75

* SEE CRD SYSTEM DESIGN SPECIFICATION.

** 2 INCH HEADER TO EACH HALF OF THE TOTAL QUANTITY OF HCU'S.

NOTES:

1. DEFINITION OF SYMBOLS
   PR = INDICATES PRESSURE OF THE REACTOR

2. MAXIMUM OPERATING TEMPERATURES
   THE MAXIMUM SYSTEM OPERATING TEMPERATURE WILL NOT EXCEED 150 DEG. F. FROM LOCATION 1 THROUGH 27 WITH THE FOLLOWING EXCEPTIONS.

	LOCATION	MAXIMUM TEMP. (DEG. F.)
MODE A -	23	200
MODE C -	23	546
	24	546
	25	280
	27	280
MODE D -	23	200
	24	280
	25	280
	27	280

3. MODE A -
   A. MAXIMUM CHARGING WATER PRESSURE SHALL BE 1600 PSIG NOMINAL, ACCUMULATOR PRECHARGE PRESSURE SHALL BE 575 PSIG NOMINAL, 500 PSIG MAXIMUM, AT 70° F.
   B. DELETED
   C. LOCATION 20 - THE CRD COOLING WATER PRESSURE SHALL NOT BE LESS THAN PR + 15 PSIG. FOR THE CONDITIONS INDICATED.
   D. LOCATION 23 - MAXIMUM DRIVE COOLING REQUIREMENTS WILL NOT EXCEED 0.34 GPM/DRIVE FOR THE CONDITIONS LISTED. MINIMUM DRIVE COOLING REQUIREMENTS WILL NOT BE LESS THAN 0.20 GPM/DRIVE.

4. MODE B -
   A. LOCATION 13 AND 14 - INSERT VALVE F007-A CLOSES ON DRIVE INSERT SIGNAL. WITHDRAW VALVE F007-B CLOSES ON DRIVE WITHDRAW SIGNAL BUT DOES NOT STAY CLOSED DURING SETTLING.
   B. LOCATION 18 - THE CRD DRIVE WATER PRESSURE SHALL NOT BE LESS THAN PR + 250 PSIG. FOR THE CONDITIONS INDICATED.

5. MODE C -
   A. DELETED
   B. THE 546 DEG. F. TEMPERATURE LISTED IN NOTE 2 FOR MODE C POSITIONS 23 AND 24 SHALL BE USED ONLY IN DETERMINING THE MINIMUM PIPE WALL THICKNESS IN VICINITY OF THE DRIVE HOUSING AND NOT IN DETERMINING STRESSES DUE TO THERMAL EXPANSION. IN DETERMINING MINIMUM WALL THICKNESS IT MAY BE ASSUMED THAT THIS TEMPERATURE OCCURS LESS THAN 1 PERCENT OF THE OPERATING LIFE OF THE SYSTEM. SEE THE CRD HYD. SYSTEM DESIGN SPECIFICATION TO DETERMINE CYCLIC STRESSES DUE TO THERMAL EXPANSION.
   C. LOCATION 21 TO 22 - THE PRESSURE DROP FROM LOCATION 21 TO 22 SHALL NOT EXCEED 475 PSI AT 30 GPM FOR ANY CRD.
   D. LOCATION 25 - A NEGATIVE FLOW RATE INDICATES FLOW FROM THE REACTOR THROUGH THE DRIVE SEAL, INTO THE CRD. THE MAXIMUM LEAK RATE FROM THE THE REACTOR CAN REACH 10 GPM PER DRIVE.
   E. LOCATION 24 TO 25 - THE PRESSURE DROP FROM LOCATION 24 TO 25 SHALL NOT EXCEED 162 PSI AT 30 GPM FOR ANY CRD.
   F. RESPONSE TIME OF FCV-F002 IS SUCH THAT SCRAM IS COMPLETED BEFORE FCV-F002 STARTS TO CLOSE.
   G. SCRAM DRAIN VALVE F011 AND VENT VALVE F010 CLOSE WITH A SCRAM SIGNAL.

6. MODE D -
   A. DELETED
   B. LOCATION 27 - THE SCRAM DISCHARGE VOLUME SHALL BE SIZED SO THAT THE RESULTING PRESSURE AFTER 100 PERCENT STROKE IS LESS THAN 65 PSIG.

7. MAXIMUM ALLOWABLE PUMP SUCTION PRESSURE SHALL BE 50 PSIG.

8. PROCESS DIAGRAM 1120I448 SHALL BE USED WITH AND FORM PART OF THIS PROCESS DATA. IF THERE ARE ANY CONFLICTS BETWEEN THE PROCESS DIAGRAM AND THIS PROCESS DATA, THE PROCESS DATA SHALL GOVERN.

9. DURING SCRAM, THIS FLOW WILL BE DIRECTED INTO THE SCRAM DISCHARGE VOLUME. FOLLOWING SCRAM, THIS FLOW WILL DECLINE AS VALVE F002 CLOSES AND AS THE SCRAM DISCHARGE VOLUME PRESSURIZES TO EQUAL THE REACTOR PRESSURE. AFTER THE SCRAM DISCHARGE VOLUME AND THE REACTOR VESSEL PRESSURE HAVE EQUALIZED, FLOW WILL BE DIVERTED TO THE REACTOR VESSEL VIA THE CRD WITHDRAW LINES AT A FLOW RATE DEPENDENT ON THE REACTOR PRESSURE:
   I.E. (A.) APPROX. 15 GPM AT "0" PSIG. REACTOR PRESSURE.
   (B.) APPROX. 6 GPM AT "1000" PSIG. REACTOR PRESSURE.

10. THIS VALUE APPLIES IMMEDIATELY FOLLOWING COMPLETION OF SCRAM. PRESSURE WILL SUBSEQUENTLY EQUALIZE WITH REACTOR PRESSURE.

11. DESIGN PRESSURE AND TEMPERATURE SHOWN IN "TABLE 1" IS FOR INFORMATION ONLY AND IS THE BASIS FOR DESIGN OF BWRS SUPPLIED EQUIPMENT. ESTIMATED LINE SIZES ARE FOR INFORMATION ONLY. ACTUAL LINE SIZES AS DETERMINED BY THE PIPING DESIGNER SHALL MEET THE PROCESS DATA HYDRAULIC REQUIREMENTS.

12. ALL VALUES SHOWN IN MODES A, B, C, AND D ARE NOMINAL UNLESS OTHERWISE NOTED.

CHKD BY

J. GARCIA

112D1448AD

Replaces FSAR
Figure 4.6-6C

NOTES:

1. DEFINITION OF SYMBOLS

   / INDICATES CONDITIONS FOR 0 FLOW RATE.

   " INDICATES THE SAME CONDITION AS LISTED UNDER MODE A.

   PR INDICATES PRESSURE OF THE REACTOR.

2. MAXIMUM OPERATING TEMPERATURES

   THE MAXIMUM SYSTEM OPERATING TEMPERATURES WILL NOT EXCEED 150 DEGREES F. FROM LOCATION 1 THROUGH 30 WITH THE FOLLOWING EXCEPTIONS.

	LOCATION	MAX TEMPERATURE (DEGREES F.)
MODE A -	23	200
MODE C -	23	546
	24	546
	25	280
	27	280
MODE D -	23	200
	24	280
	25	280
	27	280

3. MODE A -

   A. LOCATION 12 - THE RETURN LINE PRESSURE SHALL NOT EXCEED PR+50. PSIG WITH THE CRD COOLING WATER FLOW RATE AT 0.20 GPM/DRIVE. PRESSURE IN EXCESS OF PR+50 PSIG UNDER THE ABOVE CONDITIONS WILL ADVERSELY AFFECT CRD OPERATION.

   B. LOCATION 16 - THE MAXIMUM ACCUMULATOR CHARGING PRESSURE SHALL NOT EXCEED 1510 PSIG. ACCUMULATOR PRESSURE IN EXCESS OF 1510 PSIG WILL CAUSE CRD DAMAGE DURING A SCRAM.

   C. LOCATION 20 - THE CRD COOLING WATER PRESSURE SHALL NOT BE LESS THAN PR+15 PSIG FOR THE CONDITIONS INDICATED.

   D. LOCATION 23 - MAXIMUM DRIVE COOLING REQUIREMENTS WILL NOT EXCEED 0.34 GPM/DRIVE FOR THE CONDITIONS LISTED. MINIMUM DRIVE COOLING REQUIREMENTS WILL NOT BE LESS THAN 0.20 GPM/DRIVE.

4. MODE B -

   A. LOCATIONS 13 & 14 - INSERT VALVE F007-A CLOSES ON DRIVE INSERT SIGNAL. WITHDRAW VALVE F007-B CLOSED ON DRIVE WITHDRAW SIGNAL BUT DOES NOT STAY CLOSED DURING SETTLING.

   B. LOCATION 16 - THE CRD DRIVE WATER PRESSURE SHALL NOT BE LESS THAN PR+250 PSIG FOR THE CONDITIONS INDICATED.

FOR REVISIONS SEE SHEET 1

5. MODE C -

   A. CONDITIONS LISTED FOR MODE C REPRESENT THOSE CONDITIONS WHICH EXIST AT 10 PERCENT OF THE FULL STROKE INSERTION.

   B. THE 546° F. TEMPERATURE LISTED IN NOTE 2 FOR MODE C POSITIONS 23 & 24 SHALL BE USED ONLY IN DETERMINING THE MINIMUM PIPE WALL THICKNESS IN VICINITY OF THE DRIVE HOUSING AND NOT IN DETERMINING STRESSES DUE TO THERMAL EXPANSION. IN DETERMINING MINIMUM WALL THICKNESS IT MAY BE ASSUMED THAT THIS TEMPERATURE OCCURS LESS THAN 1% OF THE OPERATING LIFE OF THE SYSTEM. SEE THE CRD HYD. SYS. DESIGN SPECIFICATION TO DETERMINE CYCLIC STRESSES DUE TO THERMAL EXPANSION.

   C. LOCATIONS 21 TO 22 - THE PRESSURE DROP FROM LOCATION 21 TO 22 SHALL NOT EXCEED 436 PSI AT 90 GPM FOR ANY CRD.

   D. LOCATION 23 - A NEGATIVE FLOW RATE INDICATES FLOW FROM THE REACTOR THROUGH THE DRIVE SEAL INTO THE CRD. THE MAXIMUM LEAK RATE FROM THE REACTOR CAN REACH 10 GPM PER DRIVE.

   E. LOCATIONS 24 TO 25 - THE PRESSURE DROP FROM LOCATION 24 TO 25 SHALL NOT EXCEED 162 PSI AT 29.6 GPM FOR ANY CRD.

   F. RESPONSE TIME OF FCV-F002 IS SUCH THAT SCRAM IS COMPLETED BEFORE FCV-F002 STARTS TO CLOSE.

   G. SCRAM VENT VALVE F010 AND DRAIN VALVE F011 CLOSE WITH A SCRAM SIGNAL.

6. MODE D -

   A. LOCATIONS 24 AND 25 - A NEGATIVE FLOW RATE HERE INDICATES A TRANSIENT CONDITION IN WHICH FLOW FROM THE WITHDRAW LINE PASSES THROUGH THE CRD AND INTO THE REACTOR. DURING SCRAM THE DRIVE ACTS AS A PUMP TO CHARGE THE SCRAM DISCHARGE VOLUME TO A PRESSURE ABOVE THAT OF THE REACTOR. IMMEDIATELY FOLLOWING SCRAM, THE WITHDRAW LINE WILL REJECT WATER TO THE VESSEL UNTIL THE LOSS OF THIS WATER REDUCES THE WITHDRAW LINE PRESSURE TO APPROXIMATELY THAT OF THE REACTOR.

   B. LOCATION 27 - THE SCRAM DISCHARGE VOLUME SHALL BE SIZED SO THAT THE RESULTING PRESSURE AFTER 100% STROKE IS LESS THAN 65 PSIG.

7. PROCESS DIAGRAM 112D1448 SHALL BE USED WITH AND FORM PART OF THIS PROCESS DATA. IF THERE ARE ANY CONFLICTS BETWEEN THE PROCESS DIAGRAM AND THIS PROCESS DATA, THE PROCESS DATA SHALL GOVERN.

Delete

WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

CONTROL ROD DRIVE SYSTEM

FIGURE 4.6-6d

Delete

8. DESIGN PRESSURE & TEMPERATURE GIVEN BELOW IS FOR INFORMATION ONLY AND IS THE BASIS FOR DESIGN OF BWRS SUPPLIED EQUIPMENT. ESTIMATED LINE SIZES ARE FOR INFORMATION ONLY. ACTUAL LINE SIZES AS DETERMINED BY THE PIPING DESIGNER SHALL MEET THE PROCESS DATA HYDRAULIC REQUIREMENTS.

LOCATION	1A-1B	1B--1	2---6	3A-3B	6---9A	9A-11	7A-7B	7B-7C	11-12A	15A-15B	15B-15C
DESIGN PRESS. (PSIG)	150.	150.	1750.	1750.	1750.	1750.	1750.	1750.	1750.	1750.	1750.
DESIGN TEMP. (DEG F)	150.	150.	150.	150.	150.	150.	150.	150.	150.	150.	150.
ESTIMATED LINE SIZE (INCHES)	4.0	4.0	2.0	1.0	2.0	1.5	1.0	0.75	3.0	1.0	0.75

LOCATION	12A-12	16-16A	17-18	19-20	26-26A	21-22	24-25	27A-27B	27B-27	27-27C	27C-27D
DESIGN PRESS. (PSIG)	**	1750.	1750.	1750.	1750.	1750.	1750.	1250.	1250.	1250.	1250.
DESIGN TEMP. (DEG F.)	**	150.	150.	150.	150.	150.	150.	280.	280.	280.	280.
ESTIMATED LINE SIZE (INCHES)	3.0	2.0	1.0	***	1.0	1.0	0.75	0.75	*	10.0	2.0

*SEE CRD SYSTEM DESIGN SPECIFICATION.

**TO SAME CONDITIONS AS THE FEEDWATER PIPING (BY OTHERS)

***2" HEADER TO EACH OF THE TOTAL QUANTITY OF HCU'S.

WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

CONTROL ROD DRIVE SYSTEM

FIGURE 4.6-6e

Q. 010.045
(4.6.1.1.2.4.1)

The scram discharge volume header piping is sized to receive and contain all water discharged by the control rod drives during a scram, independent of the instrument volume. Show quantitatively how a minimum volume of 3.34 gallons per drive is required since approximately 4 gpm is required to insert the rods with up to an additional 0.34 gpm required for cooling.

Response:

The 3.34 gallon minimum volume requirement is in no way related to the 4 gpm drive insert flow rate or the 0.34 gpm cooling water flow rate. The nominal 4 gpm drive insert flow is the volumetric flow rate delivered to the underside of the control rod drive (CRD) piston to displace the piston in the upward direction and achieve normal rod insertion at about 3 inches per second. The 0.34 gpm cooling water is also delivered to the underside of the drive piston and from here it is discharged to the reactor pressure vessel (RPV) via an engineered flow path up through the annular thermal sleeve region between the CRD piston mechanism and the CRD housing. On the other hand, the scram discharge volume is sized to provide a low pressure discharge point for the volume of water above the drive piston displaced during the period of scram insertion and an additional, conservatively defined, maximum leakage from the RPV to the top of the drive piston during the scram. The volume of water over the drive piston of a fully withdrawn CRD is 0.76 gallons. To this is added a conservative 10 gpm leakage flow from the RPV for an extended period of 10 seconds. (Normal full rod insertion is complete in less than 3 seconds.) Finally, some more volume is added to accommodate the air potentially trapped in the SDV so as to assure that the SDV pressure at 10 seconds after the time of scram initiation is $\leq$65 psig The sum of the 0.76 gallons displaced from the top of the drive piston, the 10 seconds of 10 gpm post-scram leakage flow from the RPV and the free volume required for the air trapped in the SDV adds up to the specified minimum value of 3.34 gallons per CRD.

Q. 010.046
(4.6.1.1.2.4.2.2)

In Figure 4.6-5b and Drawing M528, pressure transmitter (N005) transmits a signal to a pressure switch (N600) in the process instrumentation panel in the control room, which energizes an annunciator in the control room at any time pressure in the charging header falls below the setpoint. Explain why an alarm on high is indicated for the pressure switch (N600) instead of an alarm on low which would provide protection against charging header pressure falling below the setpoint.

Response:

The charging water header of the Control Rod Drive (CRD) is monitored for high pressure since high charging water header pressure indicates the existence of an abnormal condition in the CRD hydraulic system (e.g., such as a failed close flow control valve). The pressure indicating switch on the charging water header (C11-N600) is set to actuate the control room annunciator if the charging water header pressure exceeds a nominal 1510 psig setpoint, (the alarm is actuated on an increasing pressure). Neither sustained high charging water pressure nor CRD drive water pump operation is required to successfully scram the plant. Each of the control rod drives has its own hydraulic control unit (HCU) which operates independently of any others. Scram is achieved on either HCU accumulator pressure or a combination of accumulator pressure and reactor pressure. Each HCU is safety grade and has its own accumulator. The condition of the accumulators is continuously monitored by the Reactor Manual Control System. Loss of pressure and/or leakage from any of the accumulators is detected by PSL-130 and LDS-129, respectively, for each accumulator, as shown in Figure 4.6-5b. Both occurrences are annunciated and a light signal identifies the particular scram accumulator. This instrumentation, existing locally at each HCU, provides the necessary indication of accumulator charge pressure irrespective of the pressure in the nonessential charging water header.

Q. 010.047
(4.6.1.1.2.4.2.4)

Revised FSAR Figure 4.6-5b (Amendment 16) does not appear to show valves F129, F130, F131, and F132. The applicant should explain if this is what is meant in the response by "removing the discrepancy identified above". In addition, the revised FSAR page was not provided as stated in the response. The applicant should provide the revised FSAR page and describe the revision.

Response:

FSAR Figure 4.6-5b (Amendment 16) does not show valves F129, F130, F131, and F132 because these valves were removed from the CRD system when the CRD return line to the RPV was eliminated.

What is meant by "removing the discrepancy identified above" in the response to Question 211.134 (later changed to 010.047) is that by submitting revised FSAR Figure 4.6-5b the discrepancy identified in Question 010.047 between the FSAR text, M528 (FSAR 3.2-4) and FSAR Figure 4.6-5b was resolved.

The words "revised FSAR page change attached" referred to FSAR Figure 4.6-5b.

Q. 010.048
(4.6.1.1.2.4.3.9)

The text description of the scram accumulator indicates that a check valve in the accumulator charging line prevents loss of water pressure in the event supply pressure is lost. The symbol for valve 111 in Figure 4.6-5b and Drawing M528 appears to be that of a normally open globe valve instead of a stop-check globe valve. Explain this apparent discrepancy.

Response:

There is no disagreement between 4.6.1.1.2.4.3.9 of the FSAR text and Figure 4.6-5b and Drawing M528 concerning valve 111. "A check valve in the accumulator charging line prevents loss of water pressure in the event the supply pressure is lost" refers to valve 115 and to the "charging water". Valve 111 is closed only when the pressure instrumentation is being serviced and when the nitrogen charging station is being connected and disconnected.*

*Draft FSAR page change attached.

scram valve operators. This prevents the inadvertent scram of a single drive in the event of a failure of one of the pilot valve solenoids.

4.6.1.1.2.4.3.7 Scram Inlet Valve

The scram inlet valve opens to supply pressurized water to the bottom of the drive piston. This quick opening globe valve is operated by an internal spring and system pressure. It is closed by air pressure applied to the top of its diaphragm operator. A position indicator switch on this valve energizes a light in the control room as soon as the valve starts to open.

4.6.1.1.2.4.3.8 Scram Exhaust Valve

The scram exhaust valve opens slightly before the scram inlet valve, exhausting water from above the drive piston. The exhaust valve opens faster than the inlet valve because of the higher air pressure spring setting in the valve operator.

4.6.1.1.2.4.3.9 Scram Accumulator

The scram accumulator stores sufficient energy to fully insert a control rod at lower vessel pressures. At higher vessel pressures the accumulator pressure is assisted or supplanted by reactor vessel pressure. The accumulator is a hydraulic cylinder with a free-floating piston. The piston separates the water on top from the nitrogen below. A check valve in the accumulator charging water line prevents loss of water pressure in the event supply pressure is lost.

During normal plant operation, the accumulator piston is seated at the bottom of its cylinder. Loss of nitrogen decreases the nitrogen pressure, which actuates a pressure switch and sounds an alarm in the control room.

To ensure that the accumulator is always able to produce a scram, it is continuously monitored for water leakage. A float type level switch actuates an alarm if water leaks past the piston barrier and collects in the accumulator instrumentation block.

010.049
Q. ~~211.006~~
(5.2.5)
(7.6.2)
(12.3.4)

Provide a detailed discussion of the sensitivity and response times of the containment airborne radiation monitoring systems for a number of containment background activity levels. The background activity levels which should be considered are those levels in the containment that would result from leakage through the RCPB assuming: (1) relatively clean water in the reactor coolant system at the initial operation of the WNP-2 facility at power; and (2) the maximum level of activity in the reactor coolant permitted by the WNP-2 Technical Specifications. In responding to this item, assume both the normal and the maximum leakage rates identified in your response to Question 211.115. Indicate your assumptions in estimating the response times of the containment airborne radiation monitoring systems (e.g., the preset alarm level for higher background leakage and the plateout factor).

Response:

The two types of radiation monitors used in the WPPSS Nuclear Project No. 2, for monitoring the drywell atmosphere, are the particulate monitor and the noble gas monitor. The sensitivity of these two types of monitors is given in Figure 010.049 ~~211.006~~-1 and ~~211.006-2~~. The same detector is used in both monitors. The detector's noise level is about 25 cpm. The minimum detectable concentration is based on doubling the background count rate. The count ratemeter range and the minimum sensitivity of both types of monitors are:

Noble Gas

Count Ratemeter Range:

1.4×10^{-7} μci/cc - 1.4×10^{-1} μci/cc for Kr-85

Minimum Detectable Concentration:

3.6×10^{-7} μci/cc for Kr-85

Particulate

Ratemeter Range:

2.9×10^{-12} μci/cc - 2.9×10^{-6} μci/cc for Sr/Y-90

Minimum Detectable Concentration:

7.4×10^{-12} μci/cc for Sr/Y-90

The quantity of drywell atmosphere that flows through the filter of the particulate monitor and then through the 2.2 liters chamber of the noble gas monitor is 3 cfm. The drywell atmosphere sample is returned to the drywell. (The free volume of the drywell is 200,540 ft^3). There is a charcoal filter after the particulate and before the noble gas detector. The filter efficiency of the particulate filter is assumed to be 100 percent. Similarly, the efficiency of the charcoal filter is assumed to be 100 percent for all halogens.

The 2-inch diameter scintillation crystal, of the particulate detector, is 1/4-inch away from the face of the filter tape. The filter tape is 2.5 inches wide and moves at 1"/hr.

As per the response to Question 211.005, the total minimum identifiable and unidentifiable leakage rate is taken to be 2.1 gpm. The total maximum identifiable and unidentifiable leakage rate is taken to be 5.5 gpm. The leakage is measured at drywell environmental conditions and thus the water density is assumed to be 1g/cc. Based upon potential sources of leakage (such as number and nominal size of valves), it is estimated that 38.5 percent of the leakage is due to steam leakage. Collection of the identifiable leakage is not seal-tight and thus, volatile radioisotopes can escape into the drywell atmosphere. Water leakage is assumed to be flashing, and that there is an instantaneous mixing of the volatile radioisotopes with the drywell atmosphere. It is assumed that the noble gases are in the steam phase and the small quantities of noble gases in the liquid phase are neglected. On the other hand, it is assumed that the quantities of particulate radioisotopes in the steam phase are small and thus, are neglected.

Decay was considered for the radioisotopes in the drywell and for those radioisotopes accumulated on the particulate filter. No decay was considered, however, for the radioisotopes while in transit to the detectors and while in the noble gas detector chamber (for the atmosphere in the chamber, exchange rate is 1.55 seconds). It is assumed that plating, settling, impingement, etc., reduce the specific concentration of particulate isotopes in the drywell atmosphere by a factor of 1000 before the air flow reaches the filter of the particulate monitor.

In order to be responsive to the question, the source concentrations used in this analysis were taken to be those in Table 5 of ANS/ANSI N237-1976, reduced by 1/100, as representative of "relative clean water in the reactor coolant". The design basis concentrations, as per General Electric specification document No. 22A2703F, Revision 3, were used as representative of the maximum expected level of radioactivity within the reactor coolant.

The criterion used, as an indication of leakage increase, is the doubling of the background count rate within one hour for 1 gpm (additional) leakage. Each detector was evaluated with respect to responding to the criterion. See Table ~~211.006~~-1 010.049 for the results of the analysis.

In summary, the particulate monitor will meet the requirements stated in Regulatory Guide 1.45 for the minimum activity concentration of radioisotopes in the reactor coolant. For the cases where it is assumed that the design basis activity exists in the reactor coolant, the background activity exceeds the particulate monitor range. However, the detector can be desensitized accordingly. It can be shown that if a reactor coolant activity is selected based upon the guidance contained in Regulatory Guide 1.45; i.e., if "a realistic primary coolant radioactivity concentration" is used, e.g., equal to that given in Table 5 in ANS/ANSI N237-1976, and expanding the criterion of double background count rate to the desensitized monitor, the requirements stated in the regulatory guide will be met.

The noble gas monitor, however, even though its sensitivity is consistent with Regulatory Guide 1.45 requirements, would not be capable of detecting the additional 1 gpm leakage within one hour utilizing the criterion of double background count as positive indication. The noble gas monitor does, however, provide the most reliable and fastest means of ascertaining increased activity within containment with unidentified leakages higher than 1 gpm.

TABLE ~~211.006-1~~ 010.049

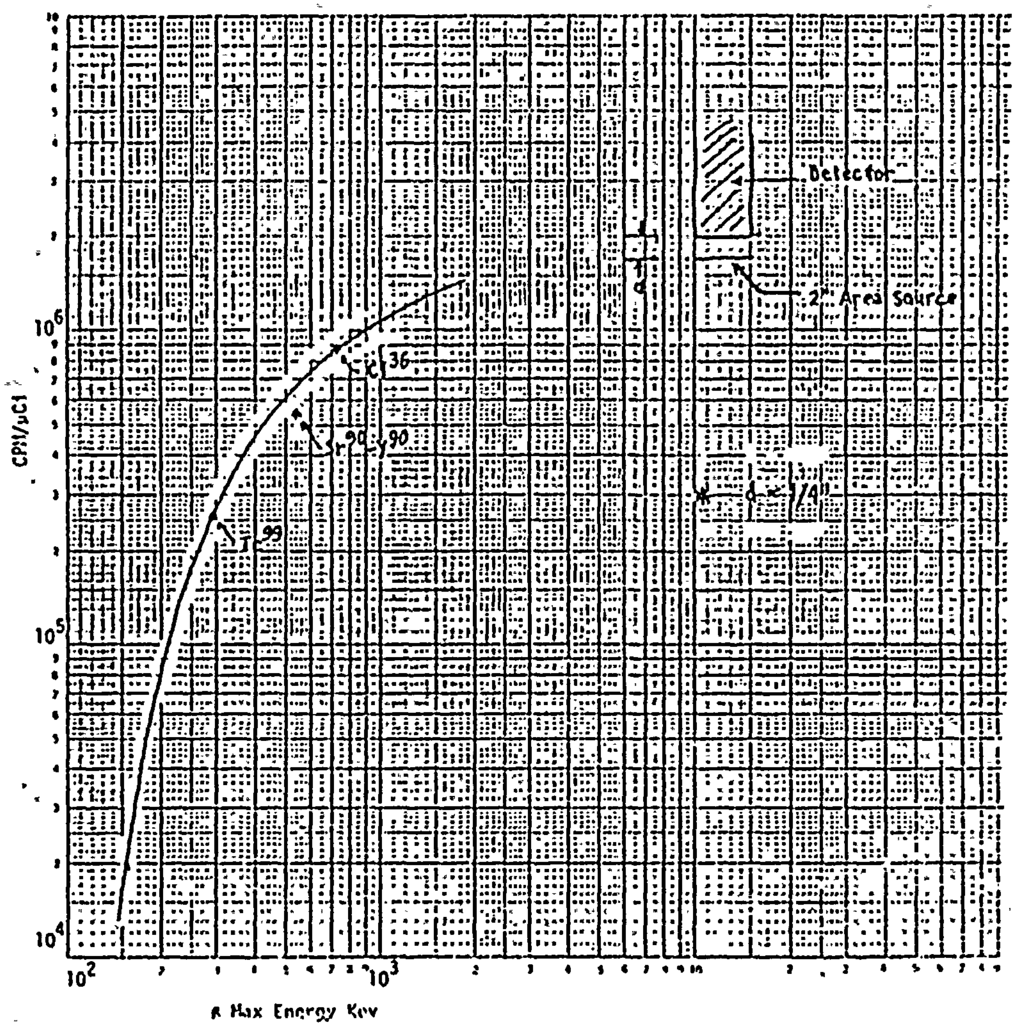
RESULTS OF MONITOR ANALYSIS

CASE	RESULTS OF ANALYSIS FOR NOBLE GAS MONITOR	RESULTS OF ANALYSIS FOR PARTICULATE MONITOR
a. Minimum activity in reactor coolant b. Minimum back-ground leak rate c. Duration of back-ground leakage = one day d. Unidentified leakage of 1 gpm is introduced.	When the background of about 5 cpm is added to the detector's noise level of ~25 cpm, the total cpm by the detector before the event is about 30. With the increase of 1 gpm leakage, the count rate would increase one hour after the event to only about 32 cpm.	The background count rate would double as a result of 1 gpm unidentified leakage within about 46 minutes.
a. Minimum activity in reactor coolant b. Maximum background leak rate c. Duration of back-ground leakage = one day d. Unidentified leakage of 1 gpm is introduced.	When the background of about 12 cpm is added to the detector's noise level of ~25 cpm, the total cpm by the detector before the event is about 37. With the increase of 1 gpm leakage, the count rate would increase, one hour after the event, to only 39 cpm.	The background count rate would double as a result of 1 gpm unidentified leakage within about 49 minutes.
a. Maximum activity in reactor coolant b. Minimum background leak rate c. Duration of back-ground leakage = one day d. Unidentified leakage of 1 gpm is introduced.	When the background of about 81 cpm is added to the detector's noise level of ~25 cpm, the total cpm by the detector, before the event, is about 106. With the increase of 1 gpm leakage, the count rate would increase, one hour after the event, to only about 136 cpm.	The background count rate exceeds the detector's sensitivity. Analysis shows that if the detector's sensitivity were high enough, it would take about 51 minutes for the background count rate to double.
a. Maximum activity in reactor coolant b. Maximum background leak rate c. Duration of back-ground leakage = one day d. Unidentified leakage of 1 gpm is introduced.	When the background of about 213 cpm is added to the detector's noise level of ~25 cpm, the total cpm by the detector, before the event, is about 238. With the increase of 1 gpm leakage, the count rate would increase, one hour after the event, to only about 269 cpm.	The background count rate exceeds the detector's sensitivity. Analysis shows that if the detector's sensitivity were high enough, it would take about 53 minutes for the background count rate to double.

010.049
TABLE ~~211.006-1~~ (Continued)

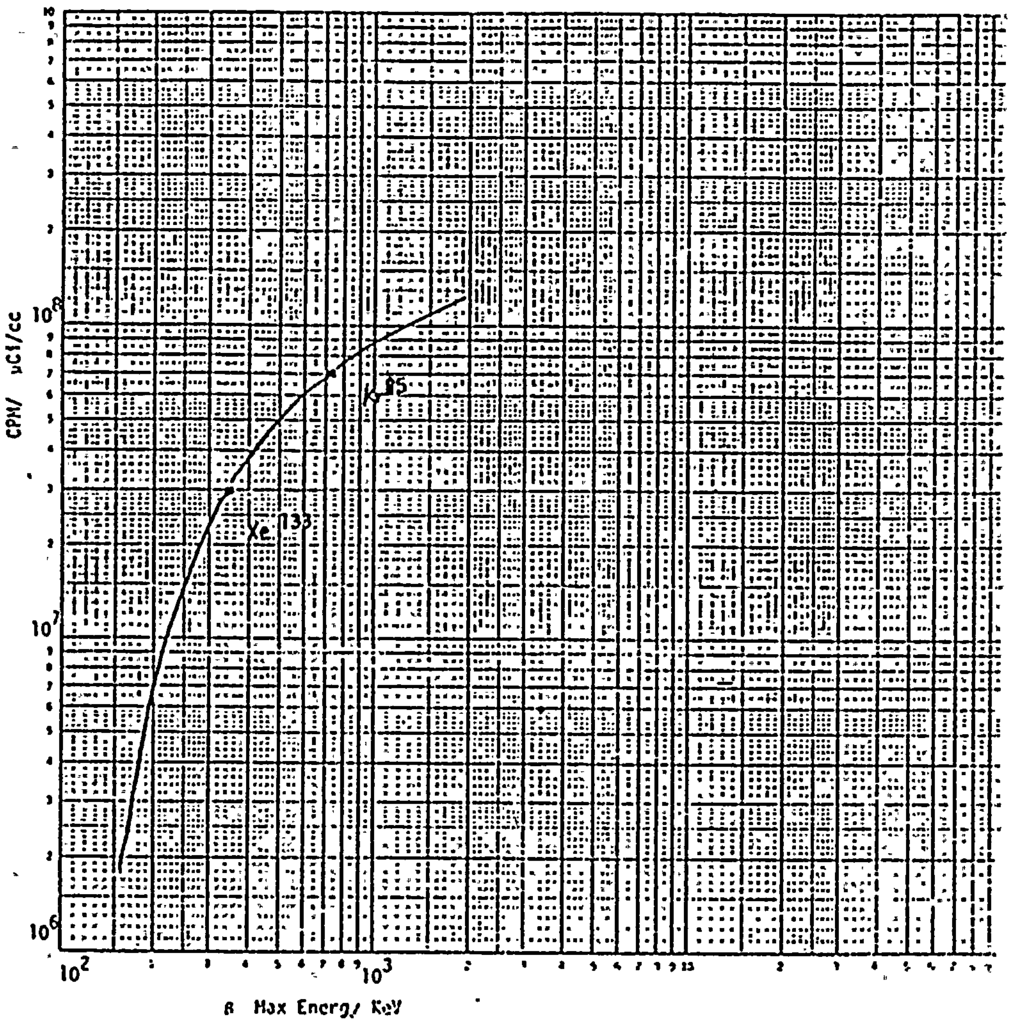
CASE	RESULTS OF ANALYSIS FOR NOBLE GAS MONITOR	RESULTS OF ANALYSIS FOR PARTICULATE MONITOR
a. Minimum activity in reactor coolant b. Minimum background leak rate c. Duration of background leakage = 100 days d. Unidentified leakage of 1 gpm is introduced.	When the background of about 10 cpm is added to the detector's noise level of ∿25 cpm, the total cpm by the detector, before the event, is about 35. With the increase of 1 gpm leakage, the count rate would increase, one hour after the event, to only about 36 cpm.	The background count rate will double as a result of 1 gpm unidentified leakage, within abut 51 minutes.
a. Minimum activity in reactor coolant b. Maximum background leak rate c. Duration of background leakage = 100 days d. Unidentified leakage of 1 gpm is introduced.	When the background of about 25 cpm is added to the detector's noise level of ∿25 cpm, the total cpm by the detector, before the event, is about 50. With the increase of 1 gpm leakage, the count rate would increase, one hour after the event, to only about 52 cpm.	The background count rate would double as a result of the 1 gpm unidentified leakage, within about 53 minutes.
a. Maximum activity in reactor coolant b. Minimum background leak rate c. Duraton of background leakage = 100 days d. Unidentified leakage of 1 gpm is introduced.	When the background of about 165 cpm is added to the detector's noise level of ∿25 cpm, the total cpm by the detector, before the event, is about 190. With the increase of 1 gpm leakage, the count rate would increase, one hour after the event, to only about 219 cpm.	The background count rate exceeds the detector's sensitivity. Analysis shows that if the detector's sensitivity were high enough, it would take about 57 minutes for the background count rate to double.
a. Maximum activity in reactor coolant b. Maximum background leak rate c. Duration of background leakage = 100 days d. Unidentified leakage of 1 gpm is introduced.	When the background of about 431 cpm is added to the detector's noise level of ∿25 cpm, the total cpm by the detector, before the event, is about 456. With the increase of 1 gpm leakage, the count rate would increase, one hour after the event, to only about 485 cpm.	The background count rate exceeds the detector's sensitivity. Analysis shows that if the detector's sensitivity were high enough, it would take about 58 minutes for the background count rate to double.

AMENDMENT NO. 17
July 1981



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	PARTICULATE SYSTEM EFFICIENCY	FIGU Att. 005-

010.0



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	GASEOUS EFFLUENT SYSTEM EFFICIENCY	FIGURE 11.006-2

010.09

Q. 010.050
(5.2.5)

The response to Question 211.007 requires additional information. It is unclear how the comparison will be made between the radioactivity monitoring and the sump level monitoring.

Describe briefly the mechanics of making these data comparisons. What calibration and operability verification tests will be performed for each independent leakage detection system? Which leakage detection system is to be used as the reference for comparison with the other systems? Do the radiation monitoring systems have radioactive sources (check sources) built into the systems?

Response:

Radiation monitors are useful as leak detection devices because of their sensitivity and rapid response to leaks. After several weeks of full power operation, a set level of background radiation will be established. Any sudden or unexplained increase in background radiation would indicate a possible primary coolant leak within the primary containment. If an increase is noted, a comparison with other leak detection devices having a relationship to each other will be made, particularly the equipment and floor drain flow rate monitors, and the reactor building sump pumps activation on high sump level. Using the flow rate monitors as a reference, the comparisons provide independent indications of a leak within the primary containment.

The radiation monitoring channels are redundant, allowing cross-checks between channels. Each channel is equipped with test sources and purge systems so that proper operation of each individual channel can be verified from the detector through to the indicator.

A discussion of calibration and operability verification tests for the leak detection system is discussed in 7.6.2.4.a.

Q. 010.051
(5.2.5)

Identified leakage is determined during preoperational testing or is measurable during reactor operation. Will the identified leakage be measured regularly and recorded? If so, provide the frequency that these data will be recorded and indicate what procedural guidelines are to be used to change the magnitude of the base identified leakage rate?

Response:

The identified leakage is measured continuously and the leakage rate will be calculated and recorded on a frequency of at least once per day in accordance with the plant technical specifications. The procedures describing how the identified leakage rate is determined will include provisions for showing the identified leakage rate numbers not exceeded the maximum allowable value of 25 gpm.

Each equipment leak-off connection has been provided with a temperature element which will identify to the operator that a higher than normal temperature exists at that particular location.

Q. 010.052
(5.2.5)

Standard Review Plan 5.2.5 specifies that unidentified leakage should be collected separately from the identified leakage so that a small unacceptable unidentified leak is not masked by larger acceptable identified leakage. Section 5.2.5 of the FSAR does not clearly indicate that separate collection of identified and unidentified leakage is provided.

Provide assurances that identified and unidentified leakage will be collected separately. If separate collection is not to be provided, provide justification for use of a common collection reservoir and show that a small unidentified leak of about 1 gpm would be recognized within one hour.

Response:

Identified and unidentified leakage are collected separately. Identified leakage is collected, monitored, and indicated by the Equipment Drain System (see FSAR Figure 3.2-9) while the unidentified leakage is collected, monitored, and indicated by the Floor Drain System (see FSAR Figure 3.2-10). Section 5.2.5.6 refers, in part, to 7.6.1.3 for further explanation. In subsections 7.6.1.3.4, 7.6.1.3.5 and 7.6.1.3.6 of this section, the two separate collection systems are described.

Q. 010.053
(5.2.5)

Provide a list of all indications available to the control room operator for evaluating and detecting unidentified leakage. Show haw the operator will determine the amount of leakage by observing the indications that are available to him, including the need for unit conversion (count rate to gpm, etc). If the monitoring is computerized, discuss the backup procedures available should the computer become inoperative.

Response:

The following indications are available to the Control Room Operator for evaluating and detecting unidentified leakage:

1.	Drywell Pressure Recorders	-3 to +3 psig 0 to 25 psig 0 to 180 psig
2.	Drywell Temperature Recorders	50° - 400° F
3.	Drywell Floor Drain Total Flow Recorder	0 - 30 GPM
4.	Reactor Building Floor Drain Sump Fillup Rate Timer	0 - 30 Min.
5.	Drywell Cooler Cooling Water Differential Temperature Recorder	0 - 150° F
6.	Reactor Vessel Water Level Recorders	-353.2" to - 153.2" -185" to +25" -35 to +85"
7.	Drywell Atmosphere Radiation Monitors	1 - 10^6 CPS

The indications listed above have no definitive correlation between their engineering units. For example, a specific count rate indicated by the Drywell Atmosphere Radiation Monitors cannot be directly converted to a leak rate in GPM, nor can a high drywell temperature be converted to an equivalent GPM leak rate. The indications listed are provided as an early warning to the operator os a potential leak. The actual unidentified leak rate is determined by observing the drywell floor drain system flow rate on flow recorders provided in the Control Room. The monitoring is not computerized.

Q. 010.056

Your response to question 010.021 is completely inadequate. your design should be modified to provide one of the following alternatives:

1. A Seismic Category I, Quality Group C, tornado missile protected Spent Fuel Pool Cooling System including the Secondary Fuel Pool Heat Exchanger Cooling System.

2. A Seismic Category I, Quality Group C, tornado mis-protected, make-up water supply to the Spent Fuel Pool and HVAC (the HVAC design environment should be 212°F and 100% humidity). The structure above the refueling floor should be seismic Category I and tornado missile protected.

3. A Seismic Category I, Quality Group C, make-up water supply to the Spent Fuel Pool and the results of an analysis which verifies that with the loss of the structure above the refueling floor, cooling with only the Seismic Category I make-up, and the most unfavorable atmospheric diffusion conditions (X/Q) that the site boundary does will not exceed 25% of the limits specified in Title 10, Code of Federal Regulations, Part 100.

Response:

A Seismic Category I, Quality Group C, tornado missile protected Spent Fuel Pool Cooling System has been provided which satisfies Alternative I. See revised Section 9.1.3.*

The valves piping, and components of the cooling portion of the Fuel Pool Cooling and Cleanup System are located within the Reactor Building and are protected form tornado generated missiles. The cooling portion of the system is Seismic Category I, and will automatically isolate from the non-seismic clean-up portion of the system on low fuel pool water level. Remote-manual startup from the main control room of redundant active components of the cooling portion of the system is provided. Safety grade cooling water to the heat exchangers and fuel pool makeup water is available from the stand-by service water (SW) system and can be initiated by remote-manual operation from the main control room. The active components of the cooling portion of the system are powered from Class IE sources.

The response to to question 010.021 has been revised to refer to this response.*

*FSAR draft page change attached.

9.1.3 SPENT FUEL POOL COOLING AND CLEANUP SYSTEM

9.1.3.1 Design Bases

The fuel pool cooling and cleanup system has been designed to comply with the objectives set forth in Regulatory Guides 1.13, Revision 1 and 1.26 Revision 3 to the extent specified in the following subsections. The system and equipment are designed to the classifications given in Tables 3.2-1 and 9.1-2.

During normal reactor operation
the system is designed to remove the decay heat released from the spent fuel elements and maintain a specified fuel pool water temperature, water clarity, and water level by accomplishing the following:

a. Minimizing corrosion product buildup and controlling fuel pool water clarity so that fuel assemblies can be efficiently handled under water.

b. Minimizing fission product concentration in the fuel pool water thereby minimizing the release of fission products from the pool to the reactor building environment.

c. Monitoring surge tank water level to thereby maintain a pool water level above the fuel sufficient to provide shielding for normal building occupancy and to control make-up flow rate from the condensate transfer system.

d. Maintaining the fuel pool water temperature below 125°F under normal operating conditions. The maximum heat load in the fuel pool under normal operating conditions occurs at the end of the 12th refueling cycle at which time there are 2068 fuel assemblies in the high density fuel racks. The estimated refueling data is given in Table 9.1-3.

* Insert

9.1.3.2 System Description

9.1.3.2.1 Normal Operation

The fuel pool cooling and cleanup system flow diagram is shown on Figure 9.1-4. System performance data are summarized in Table 9.1-1. Major components of the system are summarized in Table 9.1-2. The system is designed to dissipate the fuel pool heat load during equilibrium or non-equilibrim fuel cycle conditions.

Page 9.1-23 insert:

Following any seismic event or major plant disturbance, i.e., during abnormal operation, the system is designed to prevent fuel pool boiling and maintain adequate water level in the spent fuel pool by means of the following:

a. Automatic isolation ~~on~~ low fuel pool water level of the Seismic ~~I~~, cooling portion of the system from the non-seismic, cleanup portion of the system.

b. Remote-manual startup from the control room of redundant, active components of the fuel pool cooling portion of the system and initiation of safety grade cooling water, i.e., standby service water (SW), to the fuel pool cooling heat exchangers.

c. Remote-manual, redundant SW system make-up to the fuel pool and fuel pool level monitoring from the control room.

If required, heat removal capacity is available for the full core removal load during either of these periods, in addition to the spent fuel load already stored. The system design heat load is based on the data given in Table 9.1-3.

The system cools the fuel storage pool by transferring the spent fuel decay heat through a heat exchanger to the reactor building closed cooling water system. Water purity and clarity in the storage pool, reactor well, and dryer-separator pit are maintained by filtering and demineralizing the pool water through a filter demineralizer. In addition to fuel pool water demineralization, the system will be used on occasion to demineralize suppression pool water.

The pool cooling and cleanup system consists of two 50% capacity circulating pumps, two 50% capacity heat exchangers, two 100% filter demineralizers, two skimmer surge tanks, and the required piping, valves, and instrumentation. The pumps circulate the pool water in a closed loop, taking suction from the surge tanks, circulating the water through the heat exchangers and filters, and discharging it through diffusers at the bottom of the fuel pool and reactor well. The water flows from the pool surface through scuppers and skimmer weirs to the surge tanks. Make-up water for the system is transferred from the condensate storage tank to a skimmer surge tank to make up evaporative losses. The fuel pool pumps and heat exchangers are located in an enclosed room on the 548 foot level of the reactor building beneath the fuel pool.

Fuel pool water is continually recirculated except when draining the reactor well and dryer-separator pit. The operating temperature of 125°F is permitted to rise to 150°F when the circulating flow is interrupted for draining the reactor well and dryer-separator pit, or when larger than normal batches of fuel are stored. The fuel pool cooling and cleanup system is interconnected with the residual heat removal system to supplement the pool cooling during refueling in the event that a larger than normal batch of fuel is stored.

To establish a circulating pattern of flow in the reactor well and storage pool, the diffusers and skimmer drains are placed to sweep particles dislodged during refueling operations away from the work area and out of the pool.

Fuel pool water clarity and purity is maintained by a combination of filtering and ion exchange. The filter demineralizer maintains a total heavy element content (Fe, Cu, Hg, Ni, etc.) of 0.1 ppm or less with a pH range of 6.0 to 7.5.

Particulate material is removed from the water by the pressure precoat filter demineralizer units. The finely divided disposable filter medium is replaced when the pressure drop is excessive or the ion exchange resin is depleted. The spent filter medium is backwashed to the waste sludge phase separator tank for processing in the solid radwaste handling system. New filter medium is mixed in a precoat tank and is transferred as a slurry by a precoat pump where the solids deposit on the filter elements. The holding pump connected to each filter demineralizer maintains circulation through the filter in the interval between the precoating operation and the return to normal system operation. A strainer is provided in the effluent stream of the filter demineralizers to limit the migration of the filter material.

The two filter demineralizer units are located separately in shielded cells in the radwaste building. Sufficient clearance is provided in the cells to permit removal of the filter elements from the vessels. Each cell contains only the filter demineralizer and its associated piping. All valves are located on the outside of one shielding wall of the cell, together with necessary piping and headers, instrument elements, and controls.

Instrumentation is provided for both automatic and remote manual operation. Indication is provided in the control room and pump room. Surge tank high and low water level switches are provided. A local level indicator is provided to monitor reactor well water level. Control of flow to or from the reactor well can be accomplished during refueling. A fuel pool high/low water level switch operates a local indicator light and sounds an alarm in the control room whenever the level is either too high or too low. The trip point is adjustable over the range of the skimmer weir adjustment.

The pumps are controlled from ~~either~~ the pump room, the control room, or the vicinity of the fuel pool filters. ~~Pump low suction pressure automatically turns off the pumps~~. A pump low discharge pressure alarm annunciates in the control room and in the pump room. The controls for the remote controlled fuel pool discharge valves are located on a rack in the pump room and in the control room. The open or closed condition of each of these valves is indicated by a light in the pump room and in the control room.

The flow rate through the filter demineralizers is indicated by a flow indicator on the pump room panel.

A high rate of leakage through the refueling bellows assembly, drywell to reactor seal, or the fuel pool gates is indicated by lights on the operating floor instrument racks and is alarmed in the control room.

The filter demineralizers are controlled from a local panel. Differential pressure and conductivity instrumentation is provided for each unit to indicate when backwash is required. Suitable alarms, differential pressure indicators, and flow indicators are provided to monitor the condition of the filter demineralizers.

insert ①

9.1.3.3 Safety Evaluation

The maximum possible heat load will be the decay heat of one full core load of the fuel due to an emergency dump into the pool plus the remaining decay heat of previously discharged batches of fuel. The residual heat removal system (RHR) can be operated in parallel with the fuel pool cooling and cleanup system during this condition when the pool has a greater than normal load and when its temperature exceeds 125°F. The RHR system can be used in parallel with the fuel pool cooling system to remove abnormal heat loads, as well as during the normal refueling mode. The RHR system will not be initiated unless the reactor is in a cold shutdown condition. The operator must insert spool pieces in supply and discharge piping and open normally closed valves to permit the use of this system for supplementary cooling.

The fuel pool heat exchangers are normally cooled by the reactor building closed cooling water system to prevent contamination outside the reactor building in the event of a fuel pool heat exchanger tube failure. The system can maintain the fuel pool water temperature below 125°F when removing the nominal heat load from the pool with the reactor building closed cooling water temperature at its maximum of 95°F. The fuel pool water temperature is permitted to rise to approximately 150°F while the system water flow is diverted from the pool to drain the reactor well and dryer-separator pit, or when larger than normal batches of spent fuel are stored in the pool.

insert ②

There are no connections to the fuel storage pool which could allow the fuel pool to be drained below the pool gate between the reactor well and the fuel pool. Two diffusers are placed in both the reactor well and the fuel pool to distribute the return water as efficiently and with as little turbulance as possible. Diffusers are placed to minimize stratification of

Page 9.1-26 inserts:

(1) 9.1.3.2.2 Abnormal Operation

The portion of the FPC system which is required for cooling the fuel pool is located within the reactor building and is designed to Seismic Category I criteria. The portion of the FPC system which is used for fuel pool cleanup is located within the radwaste building and is isolable from the reactor building by means of two Seismic Category I isolation valves per line located within the reactor building. The isolation valves are either check valves or motor operated gate valves. The motor operated valves close automatically on a fuel pool low water level condition.

The redundant, active components required for fuel pool cooling are powered from division 1 and 2 power sources. Following a loss of off-site power, these components can be energized from on-site emergency power. On a loss of reactor building closed cooling (RCC) to the fuel pool heat exchangers, the RCC lines to the heat exchangers can be isolated by redundant, Seismic Category I, motor operated gate valves powered from separate Class 1E power supplies. Standby service water (SW) can be supplied to the heat exchangers through motor operated valves which are normally key locked closed. Radiation detectors are located on the SW return lines. Operation and monitoring of fuel pool cooling portion of the FPC system can be done entirely from the control room.

All components required for fuel pool cooling are qualified to the reactor building accident environment or else, they are located in enclosed rooms meeting the criteria set in 3.11.4.2. The fuel pool equipment room located on the 548 foot location is provided with both division 1 and 2 reactor building emergency cooling system.

(2) During abnormal operation the SW system is available to cool the fuel pool heat exchangers preventing any boiling of the fuel pool. The SW pressure is higher than the fuel pool pressure; thus, any leakage will be into the fuel pool system. In addition, radiation monitors on the SW return line will detect any gross tube or tube sheet failure.

either temperature or contamination. A check valve is connected to each pipe outside the pool to prevent the pool water from being siphoned out of the pool and uncovering the spent fuel. Flow control valves at the operating floor enable the operator to achieve optimum recirculation patterns to control and maintain the specified water quality and operational conditions.

*

Insert I

A make-up water valve controlled by tank level switches supplies condensate from the condensate transfer system to the pool to replace evaporative and leakage losses. The backup source of make-up water is from the Seismic Category I, Safety Class 3 standby service water system. This connection supplies makeup for long-term evaporative losses. ~~enough water to prevent the uncovering of the spent fuel. By use of the standby service water as make-up, the fuel pool will be cooled by evaporation of the pool water.~~

add * notes as shown on attachment I

~~Each~~ filter demineralizer ~~is capable of~~ continual performance, during normal operations, operating at a maximum fuel pool water flowrate of 1600 gpm ~~at a fuel pool water flow rate of 100% of rated flow~~ and will maintain water conditions as specified ~~in 9.1.3.2.~~

~~The following components of the fuel pool cooling and cleanup system (FPC) are designed to ASME Section III, Class 3: fuel pool cooling pumps, filter demineralizers, pumps, valves, and piping, FPC piping, and fuel pool heat exchanger. The system heat exchangers are also designed to the standards of the Tubular Exchangers Manufacturers Association, Class R. Piping in the reactor building is controlled and supported to Seismic Category I requirements. The water lines between the fuel pool and RHR systems are designed to ASME Section III, Class 3, Seismic Category I requirements. The FPC pumps are not designed to Seismic Category I requirements. Condensate piping in the reactor building is controlled and supported to Seismic Category I requirements.~~

A radiological evaluation of the cleanup system is presented in Chapter 12.

From the foregoing analysis, it is concluded that the fuel pool cooling and cleanup system meets its design basis and satisfies the requirements of Regulatory Guide 1.13, Revision 1 with exceptions as noted in this section.

Page 9.1-27 insert I:

All piping connecting to the fuel pool, reactor well, and dryer separator pool and their respective liner drains are Seismic Category I up to and including either the normally closed, manually operated drain valve or the normally open, redundant isolation valves which can isolate the non-seismic portion of the system. Since the fuel pool system is at low temperature and pressure (moderate energy system) postulated breaks in the Seismic Category I portion are limited to cracks.

Fuel pool cooling can be established and monitored from the control room following a design basis LOCA. Entry to the reactor building is not required. One of the two redundant trains is adequate to prevent fuel pool boiling by a large margin. Due to the large thermal capacity of the fuel pool, sufficient operator time is available after a LOCA or any other event for the operator to take action.

Attachment II for page 9.1-27.

Each filter demineralizer is capable of continual performance at a normal fuel pool water flow rate of 575 gpm or a maximum fuel pool water flow rate of 1000 gpm, and will maintain water conditions as specified in 9.1.3.2.

9.1.3.4 Testing and Inspection Requirements

No special tests except as noted below are required because at least one pump, heat exchanger, and filter demineralizer are continuously in operation while fuel is stored in the pool. Duplicate components are operated periodically to handle abnormal heat loads or to replace a unit for servicing. Routine visual inspection of the system components, instrumentation, and trouble alarms are adequate to verify system operability.

SW flow to the fuel pool heat exchangers and operability of the valves which intertie the SW and RCC systems are tested in conjunction with testing of the SW system.

TABLE 9.1-2

FUEL POOL COOLING AND CLEANUP SYSTEM EQUIPMENT DATA

Fuel Pool Heat Exchangers

Number	2
Type	Tube and Shell
Material Tube/Shell	SS/CS
Capacity, Btu/hr/heat exchanger	4.0×10^6
Cooling Water Flow, gpm/heat exchanger	575
Code and Standards	ASME/III-Class 3 and TEMA-Class R
Seismic Category	~~II~~ I

Fuel Pool Circulation Pumps

Number	2
Type	Horizontal, centrifugal
Material	SS
Flow, gpm	575
Head, Ft of H_2O	160
Motor Size hp	40
Seismic Category	~~II~~ I
Code	ASME/III-Class 3

Fuel Pool Filter Demineralizer

Number	2
Design Flow Rate, gpm	1000
Design Pressure, psig	150
Design Temperature, °F	150
Material	CS-Plastic Lined
Code	ASME/III-Class 3
Seismic ~~Class~~ Category	II

Piping and Valves

Design pressure, psig	150/300
Design Temperature, °F	220
Material	CS
Code	ASME/III-Class 3
Seismic Category	
Fuel pool cooling portion:	I
Cleanup portion:	II

Q. 010.021
(9.1.3)

Provide a cooling system and a source of makeup water for the spent fuel pool which are both designed to Seismic Category I criteria in accordance with the staff positions contained in Regulatory Guide 1.13, Revision 1, "Spent Fuel Storage Facility Design Basis," December 1975.

Response:

~~WNP-2 has a Seismic Category I source of makeup water for the spent fuel pool from the Seismic Category I standby service water service system. This is shown on Figure 9.1-4 and stated in 9.1.3.3. Cooling under emergency conditions for the fuel pool is supplied by evaporation of pool water. Regulatory Guide 1.13, Rev. 1, makes no specific statements about requiring a Seismic Category I spent fuel pool cooling system. As a result, WNP-2 meets the applicable criteria of the Regulatory Guide and the intent of the question. However, further evaluation of the design in this area is ongoing due to the interaction of fuel pool cooling and post-LOCA secondary containment pressure-temperature response. (See the response to Question 312.018).~~

See the response to question 10.56

Q. 010.057
(9.1.3)

Verify that your use of the phrase "... controlled and supported to Seismic Category I requirements" means that it meets all requirements for Seismic Category I qualification.

Response:

The cooling portion of the Fuel Pool Cooling and Cleanup System, including valves, piping, and components, meets all Seismic Category I requirements. See revised Section 9.1.3.* Non-Seismic Category I piping systems in the Reactor Building are nevertheless supported to the same Seismic Category I requirements. (See Notes 10 and 32 to Table 3.2-1.)

*Draft FSAR page change attached.

9.1.3 SPENT FUEL POOL COOLING AND CLEANUP SYSTEM

9.1.3.1 Design Bases

The fuel pool cooling and cleanup system has been designed to comply with the objectives set forth in Regulatory Guides 1.13, Revision 1 and 1.26 Revision 3 to the extent specified in the following subsections. The system and equipment are designed to the classifications given in Tables 3.2-1 and 9.1-2.

During normal reactor operation

the system is designed to remove the decay heat released from the spent fuel elements and maintain a specified fuel pool water temperature, water clarity, and water level by accomplishing the following:

a. Minimizing corrosion product buildup and controlling fuel pool water clarity so that fuel assemblies can be efficiently handled under water.

b. Minimizing fission product concentration in the fuel pool water thereby minimizing the release of fission products from the pool to the reactor building environment.

c. Monitoring surge tank water level to thereby maintain a pool water level above the fuel sufficient to provide shielding for normal building occupancy and to control make-up flow rate from the condensate transfer system.

d. Maintaining the fuel pool water temperature below 125°F under normal operating conditions. The maximum heat load in the fuel pool under normal operating conditions occurs at the end of the 12th refueling cycle at which time there are 2068 fuel assemblies in the high density fuel racks. The estimated refueling data is given in Table 9.1-3.

Insert

9.1.3.2 System Description

9.1.3.2.1 Normal Operation

The fuel pool cooling and cleanup system flow diagram is shown on Figure 9.1-4. System performance data are summarized in Table 9.1-1. Major components of the system are summarized in Table 9.1-2. The system is designed to dissipate the fuel pool heat load during equilibrium or non-equilibrim fuel cycle conditions.

Page 9.1-23 insert:

Following any seismic event or major plant disturbance, i.e., during abnormal operation, the system is designed to prevent fuel pool boiling and maintain adequate water level in the spent fuel pool by means of the following:

a. Automatic isolation on low fuel pool water level of the Seismic, cooling portion of the system from the non-seismic, cleanup portion of the system;

b. Remote-manual startup from the control room of redundant, active components of the fuel pool cooling portion of the system and initiation of safety grade cooling water, i.e., standby service water (SW), to the fuel pool heat exchangers.

c. Remote-manual, redundant SW system make-up to the fuel pool and fuel pool level monitoring from the control room.

If required, heat removal capacity is available for the full core removal load during either of these periods, in addition to the spent fuel load already stored. The system design heat load is based on the data given in Table 9.1-3.

The system cools the fuel storage pool by transferring the spent fuel decay heat through a heat exchanger to the reactor building closed cooling water system. Water purity and clarity in the storage pool, reactor well, and dryer-separator pit are maintained by filtering and demineralizing the pool water through a filter demineralizer. In addition to fuel pool water demineralization, the system will be used on occasion to demineralize suppression pool water.

The pool cooling and cleanup system consists of two 50% capacity circulating pumps, two 50% capacity heat exchangers, two 100% filter demineralizers, two skimmer surge tanks, and the required piping, valves, and instrumentation. The pumps circulate the pool water in a closed loop, taking suction from the surge tanks, circulating the water through the heat exchangers and filters, and discharging it through diffusers at the bottom of the fuel pool and reactor well. The water flows from the pool surface through scuppers and skimmer weirs to the surge tanks. Make-up water for the system is transferred from the condensate storage tank to a skimmer surge tank to make up evaporative losses. The fuel pool pumps and heat exchangers are located in an enclosed room on the 548 foot level of the reactor building beneath the fuel pool.

Fuel pool water is continually recirculated except when draining the reactor well and dryer-separator pit. The operating temperature of 125°F is permitted to rise to 150°F when the circulating flow is interrupted for draining the reactor well and dryer-separator pit, or when larger than normal batches of fuel are stored. The fuel pool cooling and cleanup system is interconnected with the residual heat removal system to supplement the pool cooling during refueling in the event that a larger than normal batch of fuel is stored.

To establish a circulating pattern of flow in the reactor well and storage pool, the diffusers and skimmer drains are placed to sweep particles dislodged during refueling operations away from the work area and out of the pool.

Fuel pool water clarity and purity is maintained by a combination of filtering and ion exchange. The filter demineralizer maintains a total heavy element content (Fe, Cu, Hg, Ni, etc.) of 0.1 ppm or less with a pH range of 6.0 to 7.5.

Particulate material is removed from the water by the pressure precoat filter demineralizer units. The finely divided disposable filter medium is replaced when the pressure drop is excessive or the ion exchange resin is depleted. The spent filter medium is backwashed to the waste sludge phase separator tank for processing in the solid radwaste handling system. New filter medium is mixed in a precoat tank and is transferred as a slurry by a precoat pump where the solids deposit on the filter elements. The holding pump connected to each filter demineralizer maintains circulation through the filter in the interval between the precoating operation and the return to normal system operation. A strainer is provided in the effluent stream of the filter demineralizers to limit the migration of the filter material.

The two filter demineralizer units are located separately in shielded cells in the radwaste building. Sufficient clearance is provided in the cells to permit removal of the filter elements from the vessels. Each cell contains only the filter demineralizer and its associated piping. All valves are located on the outside of one shielding wall of the cell, together with necessary piping and headers, instrument elements, and controls.

Instrumentation is provided for both automatic and remote manual operation. Indication is provided in the control room and pump room. Surge tank high and low water level switches are provided. A local level indicator is provided to monitor reactor well water level. Control of flow to or from the reactor well can be accomplished during refueling. A fuel pool high/low water level switch operates a local indicator light and sounds an alarm in the control room whenever the level is either too high or too low. The trip point is adjustable over the range of the skimmer weir adjustment.

The pumps are controlled from ~~either~~ the pump room, the control room, or the vicinity of the fuel pool filters. ~~Pump low suction pressure automatically turns off the pumps~~. A pump low discharge pressure alarm annunciates in the control room and in the pump room. The controls for the remote controlled fuel pool discharge valves are located on a rack in the pump room and in the control room. The open or closed condition of each of these valves is indicated by a light in the pump room and in the control room.

The flow rate through the filter demineralizers is indicated by a flow indicator on the pump room panel.

A high rate of leakage through the refueling bellows assembly, drywell to reactor seal, or the fuel pool gates is indicated by lights on the operating floor instrument racks and is alarmed in the control room.

The filter demineralizers are controlled from a local panel. Differential pressure and conductivity instrumentation is provided for each unit to indicate when backwash is required. Suitable alarms, differential pressure indicators, and flow indicators are provided to monitor the condition of the filter demineralizers.

* insert (1)

9.1.3.3 Safety Evaluation

The maximum possible heat load will be the decay heat of one full core load of the fuel due to an emergency dump into the pool plus the remaining decay heat of previously discharged batches of fuel. The residual heat removal system (RHR) can be operated in parallel with the fuel pool cooling and clean-up system during this condition when the pool has a greater than normal load and when its temperature exceeds 125°F. The RHR system can be used in parallel with the fuel pool cooling system to remove abnormal heat loads, as well as during the normal refueling mode. The RHR system will not be initiated unless the reactor is in a cold shutdown condition. The operator must insert spool pieces in supply and discharge piping and open normally closed valves to permit the use of this system for supplementary cooling.

The fuel pool heat exchangers are normally cooled by the reactor building closed cooling water system to prevent contamination outside the reactor building in the event of a fuel pool heat exchanger tube failure. The system can maintain the fuel pool water temperature below 125°F when removing the nominal heat load from the pool with the reactor building closed cooling water temperature at its maximum of 95°F. The fuel pool water temperature is permitted to rise to approximately 150°F while the system water flow is diverted from the pool to drain the reactor well and dryer-separator pit, or when larger than normal batches of spent fuel are stored in the pool.

* insert (2)

There are no connections to the fuel storage pool which could allow the fuel pool to be drained below the pool gate between the reactor well and the fuel pool. Two diffusers are placed in both the reactor well and the fuel pool to distribute the return water as efficiently and with as little turbulance as possible. Diffusers are placed to minimize stratification of

Page 9.1-26 inserts:

(1) 9.1.3.2.2 Abnormal Operation

The portion of the FPC system which is required for cooling the fuel pool is located within the reactor building and is designed to Seismic Category I criteria. The portion of the FPC system which is used for fuel pool cleanup is located within the radwaste building and is isolable from the reactor building by means of two Seismic Category I isolation valves per line located within the reactor building. The isolation valves are either check valves or motor operated gate valves. The motor operated valves close automatically on a fuel pool low water level condition.

The redundant, active components required for fuel pool cooling are powered from division 1 and 2 power sources. Following a loss of off-site power, these components can be energized from on-site emergency power. On a loss of reactor building closed cooling (RCC) to the fuel pool heat exchangers, the RCC lines to the heat exchangers can be isolated by redundant, Seismic Category I, motor operated gate valves powered from separate Class 1E power supplies. Standby service water (SW) can be supplied to the heat exchangers through motor operated valves which are normally key locked closed. Radiation detectors are located on the SW return lines. Operation and monitoring of fuel pool cooling portion of the FPC system can be done entirely from the control room.

All components required for fuel pool cooling are qualified to the reactor building accident environment or else, they are located in enclosed rooms meeting the criteria set in 3.11.4.2. The fuel pool equipment room located on the 548 foot location is provided with both division 1 and 2 reactor building emergency cooling system.

(2) During abnormal operation the SW system is available to cool the fuel pool heat exchangers preventing any boiling of the fuel pool. The SW pressure is higher than the fuel pool pressure; thus, any leakage will be into the fuel pool system. In addition, radiation monitors on the SW return line will detect any gross tube or tube sheet failure.

either temperature or contamination. A check valve is connected to each pipe outside the pool to prevent the pool water from being siphoned out of the pool and uncovering the spent fuel. Flow control valves at the operating floor enable the operator to achieve optimum recirculation patterns to control and maintain the specified water quality and operational conditions.

* Insert I

A make-up water valve controlled by tank level switches supplies condensate from the condensate transfer system to the pool to replace evaporative and leakage losses. The backup source of make-up water is from the Seismic Category I, Safety Class 3 standby service water system. This connection supplies makeup for long-term evaporative losses. ~~enough water to prevent the uncovering of the spent fuel. By use of the standby service water as make-up, the fuel pool will be cooled by evaporation of the pool water.~~

add* notes as shown on attachment II

~~Each~~ filter demineralizer is ~~capable of~~ continual performance, ~~at a fuel pool water flow rate of 100% of rated flow~~ and will ~~maintain water conditions as specified in 9.1.3.2.~~

during normal operations, operating at a maximum fuel pool water flowrate of 1000 gpm

~~The following components of the fuel pool cooling and cleanup~~ system (FPC) are designed to ASME Section III, Class 3: fuel pool cooling pumps, filter demineralizers, pumps, valves, and piping, FPC piping, and fuel pool heat exchanger. The system heat exchangers are also designed to the standards of the Tubular Exchangers Manufacturers Association, Class R. Piping in the reactor building is controlled and supported to Seismic Category I requirements. The water lines between the fuel pool and RHR systems are designed to ASME Section III, Class 3, Seismic Category I requirements. The FPC pumps are not designed to Seismic Category I requirements. Condensate piping in the reactor building is controlled and supported ~~to Seismic Category I requirements.~~

A radiological evaluation of the cleanup system is presented in Chapter 12.

From the foregoing analysis, it is concluded that the fuel pool cooling and cleanup system meets its design basis and satisfies the requirements of Regulatory Guide 1.13, Revision 1 with exceptions as noted in this section.

Page 9.1-27 insert I:

All piping connecting to the fuel pool, reactor well, and dryer separator pool and their respective liner drains are Seismic Category I up to and including either the normally closed, manually operated drain valve or the normally open, redundant isolation valves which can isolate the non-seismic portion of the system. Since the fuel pool system is at low temperature and pressure (moderate energy system) postulated breaks in the Seismic Category I portion are limited to cracks.

Fuel pool cooling can be established and monitored from the control room following a design basis LOCA. Entry to the reactor building is not required. One of the two redundant trains is adequate to prevent fuel pool boiling by a large margin. Due to the large thermal capacity of the fuel pool, sufficient operator time is available after a LOCA or any other event for the operator to take action.

Attachment II for page 9.1-27

Each filter demineralizer is capable of continual performance at a normal fuel pool water flow rate of 575 gpm or a maximum fuel pool water flow rate of 1000 gpm, and will maintain water conditions as specified in 9.1.3.2.

9.1.3.4 Testing and Inspection Requirements

No special tests except as noted below are required because at least one pump, heat exchanger, and filter demineralizer are continuously in operation while fuel is stored in the pool. Duplicate components are operated periodically to handle abnormal heat loads or to replace a unit for servicing. Routine visual inspection of the system components, instrumentation, and trouble alarms are adequate to verify system operability.

SW flow to the fuel pool heat exchangers and operability of the valves which intertie the SW and RCC systems are tested in conjunction with testing of the SW system.

TABLE 9.1-2

FUEL POOL COOLING AND CLEANUP SYSTEM EQUIPMENT DATA

Fuel Pool Heat Exchangers	
Number	2
Type	Tube and Shell
Material Tube/Shell	SS/CS
Capacity, Btu/hr/heat exchanger	4.0×10^6
Cooling Water Flow, gpm/heat exchanger	575
Code and Standards	ASME/III-Class 3 and TEMA-Class R
Seismic Category	~~II~~ I
Fuel Pool Circulation Pumps	
Number	2
Type	Horizontal, centrifugal
Material	SS
Flow, gpm	575
Head, Ft of H_2O	160
Motor Size hp	40
Seismic Category	~~II~~ I
Code	ASME/III-Class 3
Fuel Pool Filter Demineralizer	
Number	2
Design Flow Rate, gpm	1000
Design Pressure, psig	150
Design Temperature, °F	150
Material	CS-Plastic Lined
Code	ASME/III-Class 3
Seismic ~~Class~~ Category	II
Piping and Valves	
Design pressure, psig	150/300
Design Temperature, °F	220
Material	CS
Code	ASME/III-Class 3
Seismic Category	
Fuel pool cooling portion:	I
Cleanup portion:	II

Q. 010.058
(9.2.2)

Since the non-safety-related reactor building component cooling water system provides cooling for the reactor recirculation pumps, state the length of time that the pumps can be left without component cooling water flow before significant seal damage can occur, with consequent potential primary coolant leakage:

a) if pumps are kept running; and
b) if pumps are turned off.

Response:

Recirculation pump seal cooling is provided by both closed cooling water to the pump seal heat exchanger and control rod drive seal purge flow. If an event occurs where both pump seal cooling sources are lost, the pump seals will heat up, causing pump seal deterioration when temperatures exceed 250°F. Vendor test data, taken while operating at approximately 530°F and 1040psia, indicate that the seals will reach 250°F approximately 7 minutes after a total loss of cooling. This will occur whether or not the pump is running.

Similar test data indicate that if one of the two seal cooling sources is operating, the pump seal temperatures will remain below 250°F and no seal deterioration should occur.

If both pump seal cooling sources fail, resulting in extreme degradation of the pump seals, the primary coolant loss has been analyzed to be less than 70 gpm. Refer to NEDO-24083, "Recirculation Pump Shaft Seal Leakage Analysis", November 1978 (Licensing Topical Report). This small amount of primary coolant leakage will be compensated for by normal or emergency water level controls. Consequently, no hazard to the health and safety of the public will result from total loss of recirculation pump seal cooling.

The position discussed above has been presented to NRC in FSAR Appendix B, response to NUREG 0737, item II.K.3.25.

(4)

0

Q. 010.059

Regulatory Guide 1.27 requires that there be sufficient water in the spray ponds for 30 days of cooling without make-up. Discuss how you will monitor the build-up of sediment on the floor of the ponds so as to assure available of the 30 day water supply. Describe how you will clean the spray ponds without losing redundancy or degradation of the system.

Response:

Sediment build-up on the floor of the spray ponds will be monitored once every 3 months from fuel load to the first refueling outage, when the frequency of monitoring will be adjusted based on operating experience. Sediment depth will be limited to an average of 0.5 feet based on the assumption made in Section 9.2.5.3a for spray pond thermal analysis.

Sediment will be removed by sludge pumps utilizing hand held suction lines. Make-up water will be supplied by normal pond make-up. This method will allow cleaning of the spray ponds without losing redundancy or degradation of the system.

Q. 010.060

The FSAR states there is "...a suction head of at least 20 feet during RCIC operation" from the condensate storage tank at elevation 443'0" and the RCIC impeller elevation 427'3". Discuss how the 15'9" elevational difference between the condensate storage tank and the RCIC impeller satisfies the 20' requirement.

Response:

The 15'9" elevation difference stated in the question is the distance between the bottom of the condensate storage tank (CST) (el. 443'0") and the RCIC pump impeller (el. 427'3"). The RCIC pump does not take suction from the bottom of the CST, but rather from the side of the tank at elevation 445'4" (centerline of the suction pipe). The RCIC suction automatically transfers to the suppression pool when the water level in CST reaches an elevation of 447'4" (low-low-level). Therefore, the minimum static head when the RCIC pump is taking suction from the CST is at least 20 feet (el. 447.4" minus 427'3").

FSAR Section 9.2.6.5 states that at least 20 feet of "suction" head is available between the low-low setting and the RCIC pump impeller. Although this statement is correct, it has been revised to say "static head" instead of "suction head" to avoid confusion.* The suction head available to the RCIC pump when the suction transfers at low-low-level in the CST is calculated using the equation for net positive suction head (NPSH) as shown in our response to FSAR Question 022.038. Assuming a water temperature of 100°F, there is about 48 feet of NPSH available at the centerline of the pump suction which more than satisfies the 21 feet of NPSH required by the RCIC pump.

* Draft revised FSAR page change attached.

~~February 1980~~

(Revise for SCN-~~81-204~~)

The ~~E~~elevation differential~~s~~ between the low-low level setting and ~~the HPCS pump impeller (Elev. 420'-4 1/2") and~~ the RCIC pump impeller (Elev. 427'-3") provides a static ~~suction~~ head of at least 20 feet for ~~during HPCS/~~RCIC operation. ~~(Tank bottom elevation, level at 443 feet)~~ The calculated NPSH available for RCIC operation is ~~at least~~ 48 feet of which 21 feet is required.

PHM 6/25/81

Thermostatically controlled tank heaters are provided to maintain water temperature in the tanks at or above a nominal 40°F at all times. All above ground piping that contains water is heat traced to prevent freezing.

System logic diagrams are given in Chapter 7.

9.2.7 STANDBY SERVICE WATER SYSTEM

9.2.7.1 Design Bases

a. The standby service water system (SW) is designed to remove heat from plant systems which are required for a safe reactor shutdown following a LOCA.

b. The system is designed to remove reactor decay heat from the residual heat removal system during normal plant shutdown.

c. The system is designed to perform its required cooling water function following a LOCA, assuming a single active failure.

d. The system is designed to provide a means of flooding the vessel and containment, if required during the post-LOCA period.

e. The system is designed to provide makeup source of water for ensuring fuel pool evaporative cooling following a LOCA in conjunction with a design basis earthquake.

f. The system is designed to Seismic Category I and ASME Code, Section III, Class 3 requirements with the exception of that portion to and from the plant cooling towers, which is designed to ANSI B31.1 and Seismic Category II requirements.

9.2.7.2 System Description

The standby service water system includes vertical service water pumps located adjacent to the two spray ponds in two

Q. 010.061

The nitrogen bottles with its associated equipment and containment instrument air system shall be a minimum of Quality Group C.

Response:

All components of the containment instrument air system from and including the outboard containment isolation valves to the main steam isolation, safety relief, and automatic depressurization (ADS) valve operators in containment are Quality Group B. The portion of the system associated with the backup supply of nitrogen to the ADS valve operators is Quality Group C from the outboard containment isolation valves to and including the solenoid pilot valves (CIA-SPV-1A through 15A and -1B through -19B) which are mounted adjacent to the pressure reducing valves for the nitrogen bottles. See Figure 9.3.2.

The nitrogen bottles and associated control valves are standard, commercially available units that meet the requirements of Department of Transportation (COT) and Compressed Gas Association (CGA) standards. The nitrogen bottle units are mounted per Siesmic Category I, Quality Class I requirements. This type of nitrogen bottle assembly has proven highly reliable based on years of reactor operating experience.

The remaining components of the containment instrument air system, which are Quality Group D, are not essential for safe operation of the plant.

Q. 010.062
(9.4.1)

In your response to question 010.29 there seems to be a contradiction between the thickness of the air intake roof slab and the height of the top of the roof slab above grade. Please clarify your numbers and provide physical drawing(s) of the air handling system with details of the remote air intake structures.

Response:

The remote air intake structure is a buried structure. Only a portion (15") of the 24 inch thick roof slab projects above grade. Please refer to sections 4298 and 4299 of figure 3.5.52.

Q. 010.063

Discuss the control room environment which will result from the most extreme ambient and accident conditions (including the worst single failure for the HVAC). Note: The temperature/humidity for all operating/accident conditions shall be maintained within the comfort zone as defined by ASHRAE. This requirement applies to all areas which require operating personnel.

Response:

To maintain the control room at an ambient conditions which is compatible to the comfort zone defined in ASHRAE redundant seismic and environmentally qualified liquid chillers are being incorporated into the control room HVAC design. Paragraph 9.4.1 and 6.4 will be updated when this design is completed. The control room is the only area with essential equipment where personnel are routinely required during accidents.

Q. 010.064

Discuss the effects of a potential failure of the non-seismic Category I heaters in the standby service water pumphouse under the most adverse environmental conditions on the operability of the pumps.

Response:

The standby service water pumphouse electric unit heaters are designed to maintain the building above freezing during extreme environmental condition. Failure of the unit heaters during a seismic event would not degrade the operability of the pumps. If the pumps are required to operate following a seismic event, the heat generated from the pump motor is sufficient to maintain the building above freezing. Annunciation is provided in the main control room when temperature in the pump room drops to 35°F. Appropriate action can be taken from the control room such as starting of the standby service water pump to prevent pipe freezing.

Q. 010.065

Your responses to question 010.034 regarding the potential flooding of safety related equipment due to a circulating water failure are inadequate. An analysis shall be conducted in accordance with Standard Review Plant 10.4.5, "Circulating Water System", which assumes:

1) An expansion joint break (note: an incident of this type occurred at an operating BWR).

2) No credit shall be taken for isolation valve closure unless these valves are designed to safety grade requirements.

Response:

Please see revised section 10.4.5.3.*

* Draft FSAR page changes attached.

10.4.5.3 Safety Evaluation

The circulating water system is a non-safety related system. Consequently, the circulating water system is not designed to Seismic Category I requirements. Refer to 9.2.5 for a description of the ultimate heat sink which is designed to perform safety-related functions.

The condenser design assures that the pressure on the tube side is always maintained higher than the pressure on the shell side, thus eliminating leakage into the circulating water system should tube failure occur. Consequently, the design of the circulating water system precludes radioactive leakage into the system.

Periodic injection of chlorine is performed for biocide treatment, and sulfuric acid is added for scale-corrosion control within the circulating water system. An analysis of the transportation, handling, storage, and utilization of chlorine is presented in 6.4.

~~A detailed~~ Two evaluations ~~was~~ were performed to determine the effects of a postulated failure in the circulating water system inside the turbine building; a "realistic" evaluation and a boundary evaluation. For ~~this analysis~~ the "realistic" evaluation, a moderate energy crack was postulated to occur in the circulating water system barrier, (e.g., the rubber expansion joints) at the inlet to the main condenser. The inlet side was selected because it yields the severest results. For the bounding evaluation, a complete circumferential expansion joint break was assumed.

The entire condenser area is drained by means of sumps (see Figure 9.3-9), each equipped with duplex pumps. Sumps T-2 and T-3, servicing the inlet and outlet of the condenser, each have 50 gpm pumps. Each of these sumps is equipped with a level alarm and is therefore capable of detecting a circulating water system barrier failure. The level alarm will annunciate in the main control room upon reaching high level, providing a means of detecting the postulated failure within 5 minutes.

"Realistic" Break

The crack area for this postulated failure was assumed to be equal to 1/2 the pipe diameter times 1/2 the pipe wall thickness.

$$A = \frac{d}{2} \times \frac{t}{2} \quad \text{(see 3.6.2.1.4.2.b)}$$

In the first 5 minutes after a crack, 8,435 gallons of water will spill into the inlet basin. The capacity of each basin and its capability to store excess flow were calculated to be as follows:

a. Inlet basin: 22,500 gallons from El. 436 to El. 441

b. Outlet basin: 27,500 gallons from El. 436 to El. 441

c. Net volume under condenser: 180,500 gallons from El. 433 to El. 441.

The time required to fill the inlet basin, after a postulated crack occurs, is computed to be 13.3 minutes. This includes the 50 gpm outflow from the sump pump. The circulating water leakage flow will continue for 6.7 minutes after filling the inlet basin, until reaching the total estimated shutoff time of 20 minutes. It can be assumed that 10% of this water will flow out over the floor at El. 441, and the remainder, about 10,170 gallons, will flow into the condenser basin area. During this same time period, 4 sump pumps in the condenser basin area will have alternately pumped out 670 gallons, leaving 9500 gallons or 0.42 feet of water in the condenser basin. The rate of rise of water, therefore, is 0.021 ft/min during the first 20 minutes after the postulated crack occurs. Note that on high sump level, both pumps run simultaneously rather than alternately, thus doubling the calculated outflow capacity.

After the valves are closed, the water contained in the condenser unit water box will continue to discharge to the area. The quantity of water remaining is estimated to be 87,000 gallons. The flow will vary with a diminishing head, the head going from about 25 feet to zero feet. Using a 20 ft head and the same orifice flow criteria, the rate of flow will be approximately 819 gpm, discharging the remaining water in about 106 minutes. There will be an outflow from all the sump pumps of 150 gpm, with 10% of the flow from the crack again assumed to flow out over the floor. The water will accumulate in the condenser basin at about 590 gpm. After 106 minutes, the water level in this basin will rise an additional 2.77 feet, ~~at~~ AT 0.0261 ft/min. The total height of water when the discharge has stopped is therefore 3.19 feet to El. 436.19. This elevation is 5 ft. below the ~~[illegible]~~ floor level of the Turbine Bldg (el. 441), thus there will be no ~~safety~~ impacts on safety related equipment from this event.

→ [INSERT [from attached page]

~~There are no safety-ralated system components that could be affected by the flood elevation established above. Additionally, there are no safety-related electrical systems or system components that could be potentially submerged. In addition, the circulating water piping is located in a large room containing little other equipment and no safety-related equipment. Accordingly, spray effects are of no consequence. The pipes exit the rooms below grade in their routing to and from the cooling towers. Also, as stated above the floor onto which water would spill in event of a break is grade level. As a result, excess water would accumulate either in drainage basins or leak outside the building.~~

Discharge operation of water accumulated under the condenser shall be performed in accordance with radioactivity checking requirements for sump discharges.

10.4.5.4 Tests and Inspections

All system components, except the condenser, are accessible during operation and may be inspected visually. The circulating water pumps are tested in accordance with the Hydraulic Institute Standards.

Q 010.65

Insert to 10.4.5.3

Boundary Evaluation

A complete circumferential expansion joint break in the circulating water system would result in the release of large amounts of water into the turbine-generator building. The water would fill the net volume under the condensor, tripping the sump high level alarms which annunciate in the main control room. Remote-manual operation of the circulating water pumps and butterfly valves is provided in the main control room to mitigate the accident.

Disregarding operator action; however, the following evaluation is provided: Water would spill across the grade level floor of the turbine-generator building at elevation 441 ft., exiting through the railroad bay and access doors. Water could flow into the reactor building stairwells and elevator shafts from the 441 ft. elevation down to the 422 ft elevation, eventually filling the stairwells and elevator shafts with water. There is no safety-related equipment located in the stair-wells or elevator shafts. The access doors to the ECCS pump rooms at elevation 422 ft. are sealed watertight and designed to withstand a static head of 44 ft. of water. All penetrations into the reactor building below the 471 ft. elevation are sealed watertight. Water would not affect any safety related equipment in the reactor building.

Water could also spill across the grade level floor into the Radwaste /control building. The basement level of this building is 437 ft. It is thus possible to flood this level with four feet of water before the water would exit at grade level (441 ft) through access doors. There are no safety-related components which would be affected by this flooding.

The railroad bay and access doors of the turbine-generator building are not watertight and are not designed to withstand any static head of water, therefore no significant depth of water could accumulate in the turbine-generator building. All safety-related equipment in the turbine-generator building is located above the 471 ft. elevation and would not be affected.

In conclusion, a complete circumferential expansion joint break in the circulating water system inside the turbine-generator building would have no effect on safety-related equipment.

Q. 010.34
(10.4.5)

Your response to Item 010.09 is unacceptable. Specifically, your analysis of flooding due to failure of the circulating water system is based on a crack whose area is equal to one-quarter of the pipe diameter times the pipe thickness (.5t X .5d). Provide an analysis of flooding due to a postulated failure of the expansion joint in the circulating water system assuming a double-ended guillotine break at this location.

Response:

~~The original~~ response to Item 010.09 ~~has been rewritten~~ for clarity (see 10.4.5).

The double-ended guillotine break referred to above was not considered. The circulating water system is a moderate energy system by definition. Therefore, in accordance with NRC Standard Review Plan Section 3.6.1, 3.6.2, and 10.4.5, and the associated Branch Technical Position MEB 3-1, the criteria for a postulated failure shall be a through - wall leakage crack of the type addressed in the written response (10.4.5). In any case, as stated at the end of 10.4.5, circulating water piping is located remote from any safety-related equipment. The piping is located in a large room containing little other equipment and no safety-related equipment. Accordingly, safety-related equipment is not vulnerable to environmental effects of a circulating water pipe rupture. The pipe exits the room below grade in its routing to and from the cooling towers. It should be also noted that the condenser is located on grade level. Therefore, water above the floor elevation will drain outside and not collect other than in collection basins.

See revised FSAR section 10.4.5.3.

#8111090513

MECHANICAL ENGINEERING BRANCH MEETING FOR WNP-2

September 29, 30, October 1, 1981

SUMMATION OF ITEM DISPOSITION

1. Open - pending submittal of final pipe break studies
2. Closed
3. Closed
4. Closed
5. Closed
6. Closed
7. Closed
8. Closed
9. Closed - Response spectrum comparison
10. Closed
11. Closed
12. Closed
13. Closed - Computer program audit
14. Closed
15. Closed
16. Closed
17. Closed
18. Closed
19. Closed
20. Closed
21. Closed
22. Closed
23. Closed
24. Closed
25. Closed
26. Closed - Show Mark III steel containment data
27. Closed
28. Closed
29. Closed
30. Closed

SUMMATION OF ITEM DISPOSITION
(Continued)

31. Closed
32. Closed
33. Closed
34. Closed
35. Closed
36. Closed
37. Closed
38. Closed
39. Closed
40. Closed
41. Closed
42. Closed
43. Closed
44. Closed
45. Closed
46. Closed
47. Closed
48. Closed
49. Deferred to 10/9/81 meeting in NRC Offices
50. Open - pending completion of Supply System commitment.

DEFINITION OF SYMBOLS

P The Purchaser has sole responsibility.

P1 The principal responsibility lies with the Purchaser, but certain requirements which affect the safety or performance of the nuclear system will be furnished by General Electric. Design specifications and procedures prepared by the Purchaser will be made available to General Electric for review and comment.

P2 The principal responsibility lies with the Purchaser, but General Electric may review and comment on the Purchaser's detail design including test specifications and test results to determine compatibility with General Electric's requirements. In addition, General Electric may furnish information as indicated under P1, as appropriate.

GE General Electric Company has sole responsibility.

GE1 The principal responsibility lies with General Electric, but the Purchaser may furnish information on operating principles and procedures, general design guides or other plant requirements, and may review and comment.

GE2 The principal responsibility lies with General Electric, but the Purchaser may review and comment to determine basic compatibility with the Purchaser's responsibility. In addition, the Purchaser may furnish information as indicated under GE1, as appropriate.

DEFINITION OF COLUMN HEADINGS

Scope Design — This denotes responsibility for supplying pertinent design criteria and engineering data required for the design analysis and calculations to establish the essential parameters and requirements for detail equipment and plant design. This includes the preparation of outline and arrangement drawings, specifications, piping and instrumentation diagrams, process flow diagrams, functional diagrams, and heat balances.

Detail Design — This denotes responsibility for finalizing the equipment and plant design by performing the necessary analysis and calculations and by preparing the necessary drawings, specifications and instructions, such as detail construction and fabrication drawings, final arrangement and assembly drawings; purchasing, installation and testing specifications, and operating instructions.

		Scope Design	Detail Design	
I.	**REACTOR VESSEL AND INTERNALS** (Section B. 2)			
A.	Reactor vessel	GE	GE	
B.	Vessel thermal insulation	P1	P2	B&R
C.	Vessel supports	GE1	GE2	
D.	Stabilizers (vessel to sacrificial shield)	GE1	GE2	
E.	Reactor internals	GE	GE	
F.	Control rod drive housing and supports	GE	GE	
G.	In-core housing	GE	GE	
H.	Refueling bellows (vessel to containment seal)	P1	P2	B&R
I.	Instrumentation	GE	GE	
J.	Interconnecting instrument wire, cable and tubing	P1	P2	B&R
K.	Vessel pedestal and shield wall (sacrificial)	P1	P2	B&R
L.	Foundations	P1	P2	B&R
M.	Seismic analyses (reactor vessel and internals)	GE	GE	
N.	Seismic analyses (accommodation of vessel with its internals)	P1	P2	B&R
O.	Containment cooling	P1	P2	B&R
II.	**REACTOR WATER RECIRCULATION SYSTEM** (Section B. 3)			
A.	Pumps and motors	GE	GE	
B.	Pump hangers and supports	GE1	GE2	
C.	Recirculation piping and fittings	GE	GE	
D.	Piping suspension and restraints	GE1	GE2	
E.	Valves	GE	GE	
F.	Pump and piping thermal insulation	P1	P2	B&R
G.	Instrumentation	GE	GE	

		Scope Design	Detail Design	
II.	**REACTOR WATER RECIRCULATION SYSTEM (Continued)**			
H.	Interconnecting wire, cable and tubing	P1	P2	B&R
I.	Foundation and supporting steel	P1	P2	B&R
J.	Seismic analyses*	P1	P2	B&R
K.	Cooling water	P1	P2	B&R
L.	Water sampling	P1	P2	B&R
M.	Containment cooling	P1	P2	B&R
III.	**MAIN STEAM LINE** (Section B.4)			
A.	Inside drywell (from reactor vessel to second isolation valve outside primary containment)			
1.	Piping	GE1	GE2	
2.	Instrumentation	GE	GE	
3.	Interconnecting wire, cable and tubing	P1	P2	B&R
4.	Thermal insulation	P1	P2	B&R
5.	Isolation valves	GE	GE	
6.	Safety and relief valves	GE	GE	
7.	Piping suspension system	GE1	GE2	
8.	Seismic analyses*	P1	P2*	B&R.
9.	Foundations and supporting steel	P1	P2	B&R.
10.	Drain lines	P1	P2	B&R
B.	Other steam piping	P	P	B&R
C.	Seismic analyses	P1	P2	B&R.
D.	Turbine generator and steam bypass system	P1	P2	B&R.

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*GE will accommodate seismic loads resulting from the Purchaser's analyses.

		Scope Design	Detail Design	
III.	MAIN STEAM LINE (Continued)			
E.	Steam sampling	P1	P2	
F.	Plant d-c power	P1	P2	
G.	Plant air	P1	P2	
H.	Containment drywell cooling	P1	P2	
IV.	CONTROL ROD DRIVE SYSTEM (Section B.5)			
A.	Control rods	GE	GE	
B.	Control rod drives	GE	GE	
C.	Control rod drive hydraulic system			
1.	Control rod drive feed pumps and motors	GE	GE	
2.	Hydraulic control unit	GE	GE	
3.	Scram discharge header	P1	P2	B+R.
4.	Piping, hangers and valves	P1	P2	B+R
5.	Foundations and supports	P1	P2	B+R
D.	Instrumentation and readout	GE	GE	
E.	Interconnecting instrument wire, cable and tubing	P1	P2	B+R
F.	Seismic analyses*	P1	P2	
G.	Cooling water	P1	P2	
H.	Standby power	P1	P2	
I.	Plant air	P1	P2	
J.	Compartment ventilation, cooling and heating	P1	P2	
K.	Containment cooling	P1	P2	
L.	Containment accommodation	P1	P2	

*Accommodation of GE supply.

		Scope Design	Detail Design	
V.	**LOW PRESSURE CORE SPRAY SYSTEM** (Section C.2)			
A.	Pumps and motors	GE	GE	
B.	Piping, valves, hangers and thermal insulation	P1	P2	B+R
C.	Foundations and supports	P1	P2	B+R.
D.	Instrumentation	GE	GE	
E.	Interconnecting instrument wire, cable and tubing	P1	P2	B+R
F.	Seismic analyses*	P1	P2	B+R.
G.	Compartment ventilation, cooling and heating	P1	P2	
H.	Containment drywell cooling	P1	P2	
I.	Emergency equipment cooling water	P1	P2	
J.	Containment accommodation	P1	P2	
VI.	**HIGH PRESSURE CORE SPRAY SYSTEM** (Section C.3)			
A.	Pumps and motors	GE	GE	B+R.
B.	Piping, valves, hangers and thermal insulation	P1	P2	B+R
C.	Foundations and supports	P1	P2	B+R.
D.	Instrumentation	GE	GE	
E.	Interconnecting instrument wire, cable and tubing	P1	P2	B+R.
F.	Diesel generator	GE	GE	
G.	Diesel fuel storage tanks and transfer	P1	P2	B+R.
H.	Motor control center	GE	GE	
I.	Transformer	GE	GE	
J.	Metal clad switchgear	GE	GE	

*Accommodation of GE supply.

		Scope Design	Detail Design	
VI.	HIGH PRESSURE CORE SPRAY SYSTEM (Continued)			
K.	Seismic analyses*	P1	P2	B&R
L.	Standby power	P1	P2	
M.	Compartment ventilation, cooling and heating	P1	P2	
N.	Containment drywell cooling	P1	P2	
O.	Emergency equipment cooling water	P1	P2	
P.	Containment accommodation	P1	P2	
VII.	RESIDUAL HEAT REMOVAL SYSTEM (Section C. 4)			
A.	Heat exchangers	GE1	GE	
B.	Pumps and motors	GE	GE	
C.	Service water pumps and motors	P1	P2	B&R
D.	Piping, valves, hangers and thermal insulation	P1	P2	B&R
E.	Foundations and supports	P1	P2	B&R
F.	Instrumentation (less service water instrumentation)	GE	GE	
G.	Interconnecting instrument wire, cable and tubing	P1	P2	B&R
H.	Instrumentation for service water supply	P1	P2	
I.	Seismic analyses*	P1	P2	
J.	Cooling water	P1	P2	
K.	Standby power	P1	P2	
L.	Compartment ventilation, cooling and heating	P1	P2	
M.	Containment cooling	P1	P2	
N.	Emergency equipment cooling water	P1	P2	
O.	Containment accommodation	P1	P2	

*Accommodation of GE supply.

		Scope Design	Detail Design	
VIII.	REACTOR CORE ISOLATION COOLING SYSTEM (Section C. 5)			
A.	Pump and turbine drive	GE	GE	
B.	Piping, valves, hangers and thermal insulation	P1	P2	B&R.
C.	Foundations and supports	P1	P2	B&R
D.	Instrumentation	GE	GE	
E.	Interconnecting instrument wire, cable and tubing	P1	P2	B&R
F.	Seismic analyses*	P1	P2	
G.	Plant d-c power	P1	P2	
H.	Compartment ventilation, cooling and heating	P1	P2	
I.	Containment cooling	P1	P2	
J.	Makeup water volume	P1	P2	
K.	Emergency equipment cooling volume	P1	P2	
L.	Containment accommodation	P1	P2	
IX.	STANDBY LIQUID CONTROL (Section C. 6)			
A.	Storage tank	GE	GE	
B.	Test tank	GE	GE	
C.	Pumps and motors	GE	GE	
D.	Piping, hangers and thermal insulation	P1	P2	B&R
E.	Explosive valves	GE	GE	
F.	All other valves	P1	P2	B&R.
G.	Foundations	P1	P2	B&R.
H.	Instrumentation	GE	GE	
I.	Interconnecting instrument wire, cable and tubing	P1	P2	B&R

*Accommodation of GE supply.

			Scope Design	Detail Design	
IX.	STANDBY LIQUID CONTROL (Continued)				
	J.	Sodium pentaborate solution	P1	P2	B&R
	K.	Seismic analyses*	P1	P2	
	L.	Water quality	P1	P2	
	M.	Containment accommodations	P1	P2	
X.	REACTOR WATER CLEANUP SYSTEM (Section C. 7)				
	A.	Filter/demineralizer (including filter/ demineralizer units, precoat tank, holding pump, precoat pump agitators, post strainer and valves)	GE	GE	
	B.	Regenerative heat exchanger	GE	GE	
	C.	Nonregenerative heat exchanger	GE1	GE	
	D.	Cleanup recirculating pumps and motors	GE	GE	
	E.	Piping, valves, hangers and thermal insulation	P1	P2	B&R
	F.	Foundations and supports	P1	P2	"
	G.	Instrumentation	GE	GE	
	H.	Interconnecting instrument wire, cable and tubing	P1	P2	B&R
	I.	All resins	P1	P2	B&R
	J.	Seismic analyses*	P	P	B&R
	K.	Cooling water	P1	P2	
	L.	Standby power	P1	P2	
	M.	Containment cooling	P1	P2	
	N.	Makeup water volume	P1	P2	
	O.	Containment accommodation	P1	P2	

*Accommodation of GE supply.

WPPSS
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		Scope Design	Detail Design	
XI.	OFFGAS SYSTEM - AIR EJECTOR (Section C.8)			
A.	Pumps and motors	GE	GE	
B.	Process equipment	GE	GE	
C.	Filter holders and elements	P1	P2	B+R.
D.	Piping, valves, hangers and thermal insulation	P1	P2	B+R.
E.	Shielded vault	P1	P2	B+R.
F.	Instrumentation	GE	GE	
G.	Interconnecting instrument wire, cable and tubing	P1	P2	B+R
H.	Seismic analyses*	P1	P	
I.	All others	P	P	
XII.	RADWASTE DISPOSAL SYSTEM (Section C.9)			
A.	Tanks at radwaste facility			
	1. Filter aid tank and agitator	GE	GE	
	2. Precoat tank and agitator	GE	GE	
	3. Cleanup phase separator	GE	GE	
	4. Feed concentrate tank	P	P	7
	5. Waste sludge phase separator	GE 1	GE	26, 36
	6. All others	P1	P	
B.	Floor drain filter and equipment	GE	GE	
C.	Waste collector filter and equipment	GE	GE	
D.	Waste collector and floor drain demineralizer and equipment	GE	GE	
E.	Sump heat exchangers	P1	P2	
F.	Resins	P1	P2	
G.	Decontamination solution waste concentrator and equipment	GE	GE	
H.	Mixer	P	P	7

* Accommodation of GE supply.
Revised 3/20/72 (Reference C.O. No. 7)
Revised 4/29/74 (Reference C.O. No. 26)
Revised 1/13/75 (Reference C.O. No. 36)

		Scope Design	Detail Design	
XII.	**RADWASTE DISPOSAL SYSTEM** (Cont.)			
I.	Centrifuge	GE	GE	
J.	Conveyor system	P	P	7
K.	Hydraulic press	P1	P2	
L.	Detergent drain filter	GE	GE	
M.	Surge hoppers and equipment	P	P	7
N.	Condensate phase separator	GE	GE	
O.	Instrumentation	GE	GE	
P.	Interconnecting instrument wire, cable and tubing	P1	P2	
Q.	Control Valves	GE	GE	
R.	Piping, valves, hangers and thermal insulation	P1	P2	
S.	Foundations	P1	P2	
T.	Local panels and racks	GE	GE	
U.	Seismic analyses*	P1	P	
V.	Cooling water	P1	P2	
W.	Makeup water volume	P1	P2	
XIII.	**TOOLS AND SERVICING EQUIPMENT** (Section D)			
A.	Fuel servicing	GE	GE	
B.	Servicing aids	GE	GE	
C.	Reactor vessel servicing	GE	GE	
D.	In-vessel servicing	GE	GE	
E.	Refueling ~~equipment~~	GE	GE	
F.	Storage equipment	GE	GE	
G.	Under-vessel servicing	GE	GE	
H.	All others	P	P	
I.	Seismic analyses*	P1	P	

* Accommodation of GE supply.

Revised 3/20/72 (Reference C.O. No. 7)

		Scope Design	Detail Design
XIV.	NEUTRON MONITORING SYSTEM (Section E.2)		
A.	Sensors and signal conditioning instruments	GE	GE
B.	Sensor drive systems and tubing	GE	GE
C.	Interconnecting instrument wire and cable	P1	P2
D.	Seismic analyses*	P1	P2
E.	Plant d-c power	P1	P2
F.	24-volt d-c system	P1	P2
G.	Containment cooling	P1	P2
H.	Containment accommodation	P1	P2
XV.	REACTOR PROTECTION SYSTEM (Section E.3)		
A.	Sensors and logic equipment	GE	GE
B.	Protection system M-G set	GE	GE
C.	Interconnecting instrument wire, cable and tubing	P1	P2
D.	Seismic analyses*	P1	P2
E.	Standby power	P1	P2
F.	Containment cooling	P1	P2
XVI.	FEEDWATER CONTROL SYSTEM (Section E.4)		
A.	Flow nozzles	GE1	GE
B.	Interconnecting instrument wire, cable and tubing	P1	P2
C.	Instrumentation	GE1	GE
D.	Feedwater system (pumps, heaters and piping)	P1	P2
E.	Seismic analyses*	P1	P

*Accommodation of GE supply.

WPPSS
12/1/0

		Scope Design	Detail Design
XVI.	FEEDWATER CONTROL SYSTEM (Section E. 4) (Continued)		
	F. Feedwater sampling	P1	P2
	G. Plant air	P1	P2
	H. Containment cooling	P1	P2
XVII.	MAIN STEAM LINE RADIATION MONITORING (Section E. 5)		
	A. Main steam line radiation monitors	GE	GE
	B. Interconnecting instrument wire, cable and tubing	P1	P2
	C. Seismic analyses*	P1	P
	D. 24-volt d-c system	P1	P2
	E. All others	P	P
XVIII.	COMPUTER — NUCLEAR STEAM SUPPLY SYSTEM (Section E. 6)		
	A. Equipment	GE1	GE
	B. Interconnecting instrument wire and cable	P1	P2
	C. Programming	GE1	GE
	D. Seismic analyses*	P1	P
XIX.	PANELS, CABINETS AND RACKS (Section E. 7)		
	A. Control and auxiliary equipment room panels		
	1. Nuclear system	GE1	GE
	2. Interconnecting instrument wire, cable and tubing (external to panels)	P1	P2
	3. Compartment ventilation, cooling and heating	P1	P2

*Accommodation of GE supply.

		Scope Design	Detail Design
XIX.	**PANELS, CABINETS AND RACKS** (Continued)		
	4. All others	P1	P2
B.	Local panels and racks		
	1. Nuclear system	GE	GE
	2. Interconnecting instrument wire, cable and tubing (external to panels)	P1	P2
	3. All others	P	P
C.	Seismic analyses*	P1	P2
XX.	**PROCESS RADIATION INSTRUMENTATION (OPTIONAL)** (Section G. 1)		
A.	Elevated release point exhaust monitors	GE	GE
B.	Offgas radiation monitors	GE	GE
C.	Liquid process radiation monitors	GE	GE
D.	Reactor building vent plus exhaust plenum	GE	GE
E.	Building ventilation exhaust sampler	GE	GE
F.	Carbon bed vault radiation monitor	GE	GE
G.	Interconnecting instrument wire, cable and tubing	P1	P2
H.	Seismic analyses*	P1	P
I.	24-volt d-c system	P1	P2
J.	All others	P	P
XXI.	**AREA RADIATION MONITORING** (OPTIONAL) (Section G. 2)		
A.	Instrumentation	GE	GE
B.	Panel	GE	GE
C.	Interconnecting wire and cable	P	P

*Accommodation of GE supply.

		Scope Design	Detail Design
XXII.	ENVIRONS MONITORING (OPTIONAL) (Section G.3)		
A.	Instrumentation	GE	GE
B.	Panel	GE	GE
C.	Interconnecting wire and cables	P	P
D.	Station structures	P	P
XXIII.	COMPUTER – BALANCE-OF-PLANT (OPTIONAL) (Section G.5)		
A.	Equipment, except Turbine Sequence Monitor System	GE1	GE 24
1.	Turbine Sequence Monitor System	P	P
B.	Interconnecting instrument wire and cable	P1	P2
C.	Programming	GE1	GE
XXIV.	OTHER PLANT SYSTEMS AND EQUIPMENT	P1	P

Revised 4/22/74 (Reference C.O. No. 24)

3.6.2 Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

Question 1

In order for us to complete our review, the applicant should provide a summary of the data developed to select postulated break locations including, for each point, the calculated stress intensity, the calculated cumulative usage factor, and the calculated primary plus secondary stress range. This data is required for review to ensure that the pipe break criteria have been properly implemented. Figures 3.6-11 through 3.6-36 are not completed.

Response:

[illegible]

Postulated pipe break locations were selected on a basis of significant change in flexibility in high energy piping systems. Examples of change in flexibility are pipe fittings (elbows, tees and reducers) and circumferential connections to valves and flanges. This method of selection was chosen since it was conservative and most expedient, not requiring the availability of detailed stress analysis of the piping systems. The use of this criteria could only result in the need for too many pipe whip restraints rather than too few. In cases where the design and installation of appropriate pipe whip restraints might prove to be difficult, an option always remained to evaluate the need for the restraint on the basis of stress criteria when the final stress analysis became available. The significance of the use of flexibility criteria for postulated pipe break locations is that it permitted the design of pipe whip restraints (if required) at an early date in the project.

Following selection of postulated break points, it was necessary to determine the movement of the pipe due to jet reaction. It was not always necessary to provide a pipe whip restraint for every postulated break. Where it was determined that the movement of the pipe did not strike any piping system and/or equipment necessary for safe shutdown of the reactor or necessary to mitigate the consequences of a LOCA and where the whipping pipe would not directly strike the primary containment vessel, then pipe whip restraint was not required. Where pipe whip restraint was required, the pipe whip restraint was designed to meet the unique conditions for the postulated break.

As indicated above, stress criteria referred to in the question did not enter into the establishment of the basic design for WNP-2. However, additional studies are now underway wherein stress criteria for determination of break locations are being applied. Stress-related information used in connection with these studies will be provided when it becomes available upon the completion of these studies.

Summation - In connection with the additional studies discussed in paragraph 3 of the response, the results of these studies will be supplied in the FSAR together with specific stress criteria to complete Figures 3.6-11 through 3.6-36.

3.6.2 Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

Question 2

It is the staff's position that breaks must be postulated at any location where the cumulative usage factor exceeds 0.1. At these locations, both circumferential and longitudinal pipe breaks should be postulated, unless it can be clearly shown that the high usage factor is due primarily to stresses in only one principle direction. The applicant's response to Q. 110.012 states that the rules set forth in 3.6.2.1.4.1e (1) and (2) exempt certain break orientations based solely on stress and are independent of calculated cumulative usage factor. Clarification of this area is required.

RESPONSE

Where cumulative usage factor is a determinant in establishing a postulated break location, then to determine whether both a circumferential and longitudinal break need be postulated, the stresses in the two directions are compared.

FSAR page change (3.6-30a) agreed.

> "(1) If the result of a detailed stress analysis indicates that the maximum stress range in the axial direction is at least 1.5 times that in the circumferential direction, only a circumferential break is postulated. Where usage factor is a determinant in establishing a postulated break location, the fatigue dominant stresses are examined as indicated above to determine whether longitudinal, circumferential or both are postulated"

Summation - This item is closed.

(1) If the result of a detailed stress analysis indicates that the maximum stress range in the axial direction is at least 1.5 times that in the circumferential direction, only a circumferential break is postulated. Where usage factor is a determinant in establishing a postulated break location, the fatigue dominant stresses are examined as indicated above to determine whether longitudinal, circumferential or both are postulated.

MEB Question 2

3.6.2 Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

Question 3

For ASME, Section III, Class 1 piping designed to Seismic Category I standards, breaks due to stress are to be postulated at the following locations:

a. If Eq. (10), as calculated by Paragraph NB-3653, ASME Code Section III, exceeds 2.4 Sm, then Eqs. (12) and (13) must be evaluated. If either Eq. (12) or (13) exceeds 2.4 Sm, a break must be postulated. In other words, a break is postulated if

$$\text{Eq. } (10) > 2.4\ S_m \text{ and Eq. } (12) > 2.4\ S_m$$
or
$$\text{Eq. } (10) > 2.4\ S_m \text{ and Eq. } (13) > 2.4\ S_m$$

b. Breaks must also be postulated at any location where the cumulative usage factor exceeds 0.1.

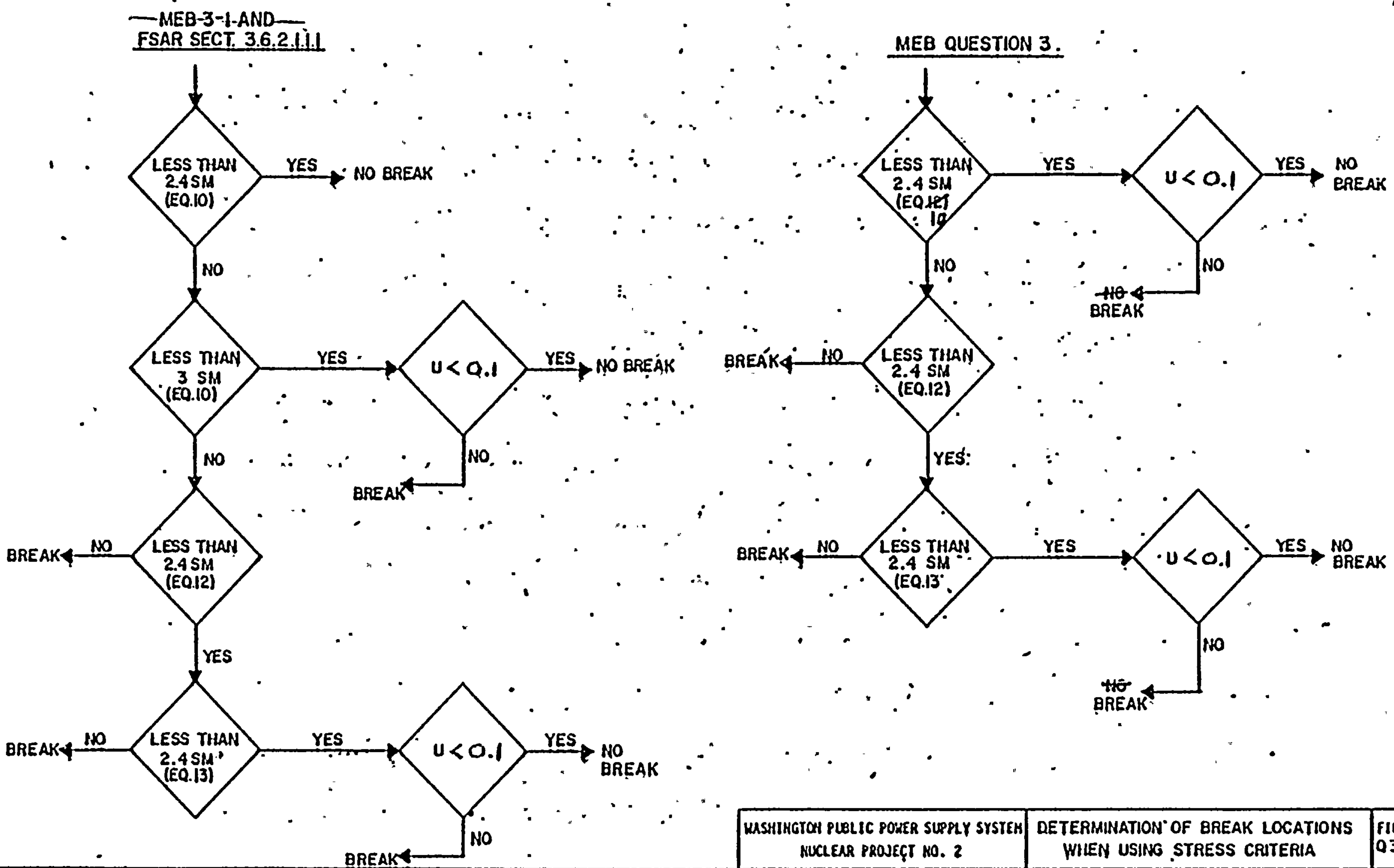
The above criteria is evaluated under loadings resulting from normal and upset plant conditions including the OBE. Any deviations from the above criteria must be justified.

Response:

WNP-2 will determine break locations for ASME, Section III, Class 1 piping consistent with the NRC position given in Branch Technical Position MEB 3-1.

Summation - This item is closed.

ASME SECTION III, SS I PIPING



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	DETERMINATION OF BREAK LOCATIONS WHEN USING STRESS CRITERIA	FIGURE Q3-1

Note: SM = S_m

[illegible]

3.6.2 Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

Question 4

A. For those portions of ASME Section III, Class 1 piping discussed in FSAR Section 3.6.2.1.2.1 and seismic Category I standards and included in the break exclusion area, breaks need not be postulated providing all of the following criteria are met:

a. Eq. (10) as calculated by Paragraph NB-3653, ASME Code, Section III, does not exceed 2.4 Sm.

b. If Eq. (10) does exceed 2.4 Sm, then Eqs. (12) and (13) must be evaluated. If neither Eq. (12) or (13) exceeds 2.4 Sm, a break need not be postulated. In other words, a break need not be postulated if:

Eq. (10) $> 2.4\ S_m$ and Eq. (12) $< 2.4\ S_m$ and

Eq. (13) $< 2.4\ S_m$

c. The cumulative fatigue usage factor is less than 0.1.

d. For plants with isolation valves inside containment, the maximum stress, as calculated by Eq. (9) in ASME Code Section III, Paragraph NB-3552 under the loadings of internal pressure, deadweight and a postulated piping failure of fluid systems upstream or downstream of the containment penetration area must not exceed 2.25 S_m.

The above criteria is evaluated under loadings resulting from normal and upset plant conditions including OBE.

In addition, augmented inservice inspection is required on all ASME Class 1, 2 and 3 piping in the break exclusion area. It is not clear whether footnote (a) on page 3.6-28 of the

4)

FSAR is applicable to Section 3.6.2.1.2.2. The applicant must provide assurances that their criteria for piping in the break exclusion areas complies with the requirements outlined above and those of Standard Review Plan 3.6.3. A list of all systems included in the break exclusion areas must be included in the FSAR. In addition, break exclusion areas should be shown on the appropriate piping drawings.

B. a. Document the method used to verify that the stresses in welded flued head fittings meets the limits specified in MEB 3-1. Indicate on which piping system the welded flued heads are used.

b. Describe the inservice inspection of the welded flued head fittings.

Response:

A. Footnote (a) on page 3.6-28 is not applicable to 3.6.2.1.2.2. The revision to page 3.6-28 indicates the systems with break exclusion areas between primary containment isolation valves. These systems are ASME Section III Class 1 systems.

---

*With respect to loadings resulting from postulated piping failure outside the exclusion area (Section B.1.b.(1)(d) of MEB 3-1) detailed analyses were performed for the main steam and feedwater systems for breaks inside and outside of containment. These analyses have confirmed the acceptability of main steam and feedwater systems and have been used to conclude that detailed analyses for smaller lines are not required.

Please refer to revised FSAR page 3.6-28 for a list of all systems included in the break exclusion area. See Figures 3.6-147a through 3.6-147e for break exclusion areas.

B.a. A project unique analysis of the welded flued head fitting in the mainsteam penetration has been performed to assure meeting the code stress limits and the NRC criteria for the break exclusion area and documented in the WNP-2 Stress Report. In addition, GE has demonstrated the physical integrity of the WNP-2 flued head by a bounding generic finite element analysis on a similar configuration; this is documented in the GE Report NEDO-23652.

b. The flued head to process pipe weld is examined volumetrically from axial and perpendicular surfaces using ultrasonic methods. By using 45 and 60 degree shear wave from the perpendicular surface and 0 degree longitudinal wave from the axial surface complete coverage of the weld is assured. Details of the examination are contained in procedure UTP-33, which is contained in the WNP-2 PSI Program Plan. In addition, surface examination will be performed on the accessible portion of the weld.

Summation - This item is closed.

3.6.2.1.2.1 Postulated Pipe Break Locations in ASME Section III Class I Piping Between Primary Containment Isolation Valves

No pipe breaks are postulated in the portion of piping between primary containment isolation valves, if any of the following apply:

(1) S_n does not exceed $2.4S_m$.

(2) S_n exceeds $2.4S_m$ but does not exceed $3S_m$, and the Cumulative Usage Factor (U) does not exceed 0.1.

(3) S_n exceeds $3S_m$, but S_e and S_r are each less than $2.4S_m$, and U does not exceed 0.1.

The stress levels in the ASME Section III Class I containment penetration high energy piping are maintained at or below these limits and therefore, breaks are not postulated.[a] See 3.6.2.1.2.3 for further discussion of containment penetration piping.

A Insert attached

3.6.2.1.2.2 Postulated Pipe Break Locations in ASME Section III Class 2 and 3 Piping Between Primary Containment Isolation Valves

See 3.6.2.1.1.2 b. (2) for stress criteria applicable to ASME Section III Class 2 and 3 piping between containment isolation valves.

The stress levels are maintained at or below these limits and therefore breaks are not postulated. See 3.6.2.1.2.3 for further discussion of containment penetration piping.

3.6.2.1.2.3 Primary Containment Penetration Piping

Primary containment penetrations, in order to maintain containment integrity, are designed with the following characteristics:

(a) A program for augmented inservice inspection will be included in the WNP-2 Inservice Inspection Program Plans to provide one hundred percent volumetric examination; each inspection interval of all pressure boundary welds in Class I high energy piping exceeding one inch nominal diameter between containment isolation valves for which no breaks are postulated.

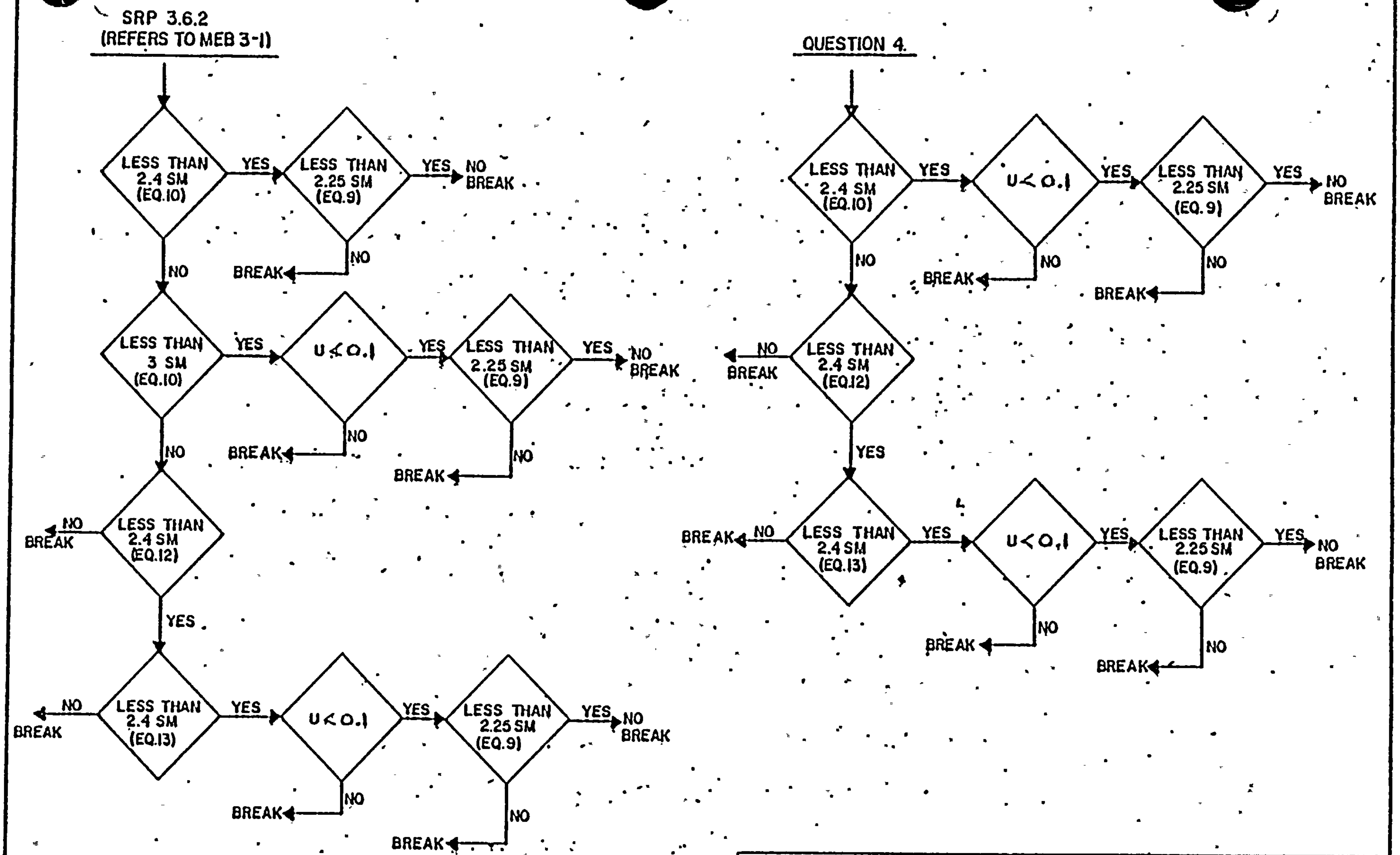
Insert to Page 3.6-28:

Piping systems which may have break exclusion areas between primary containment isolation valves are those determined by examining the list of high energy piping systems (see 3.6.2.1 and Table 3.6-2). Systems which do not pass through primary containment are excluded. In addition, systems which are not pressurized between the isolation valves during normal plant operation (see 3.6.2) are excluded. The remaining systems, those which may have break exclusion areas between primary containment isolation valves, are listed in Table 3.6-18. Break exclusion areas for these systems are shown on Figures 3.6-147a through 3.6-147e.

TABLE 3.6-18

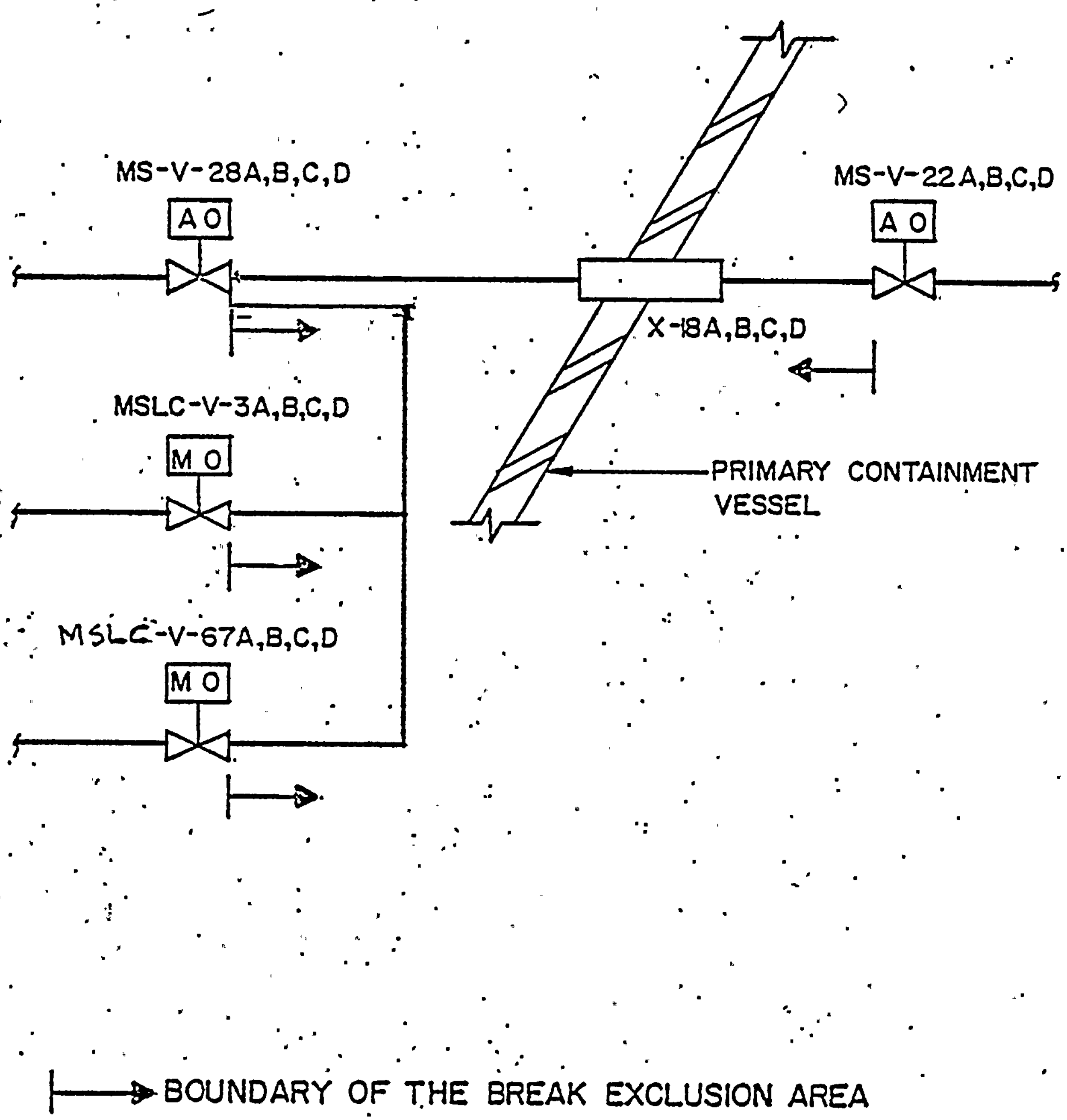
PIPING SYSTEMS CONTAINING BREAK EXCLUSION AREAS BETWEEN PRIMARY CONTAINMENT ISOLATION VALVES

PIPING SYSTEM	PIPE SIZE
Main Steam Loop A	26"
Main Steam Loop B	26"
Main Steam Loop C	26"
Main Steam Loop D	26"
Reactor Feedwater Line A	24"
Reactor Feedwater Line B	24"
RHR Condensing Mode/ RCIC Turbine Steam	10"/4"
Reactor Water Cleanup	6"
~~Main Steam Valves Drainage Piping~~	~~3"~~

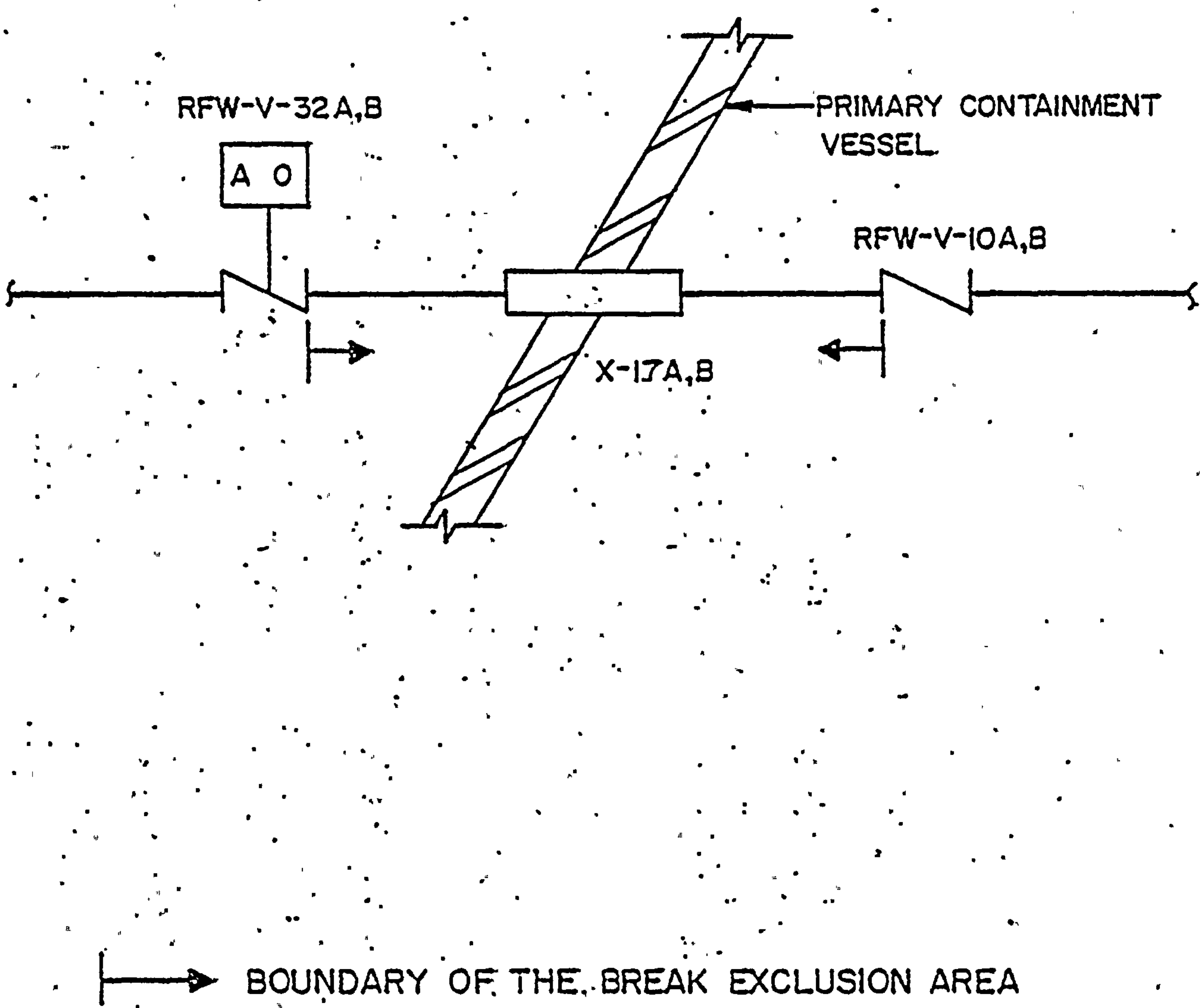


WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	COMPARISON OF BREAK LOCATION CRITERIA	FIGURE Q4-1

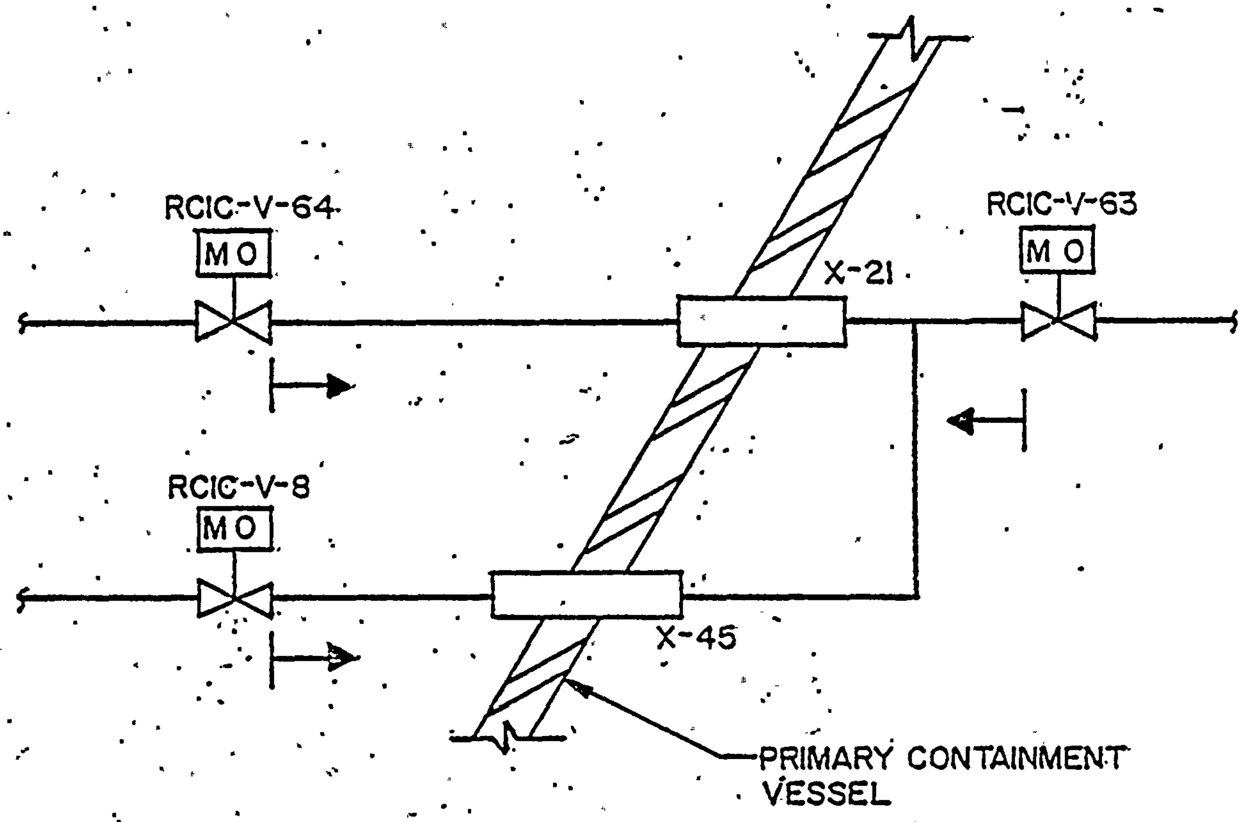
Note: SM = Sm



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	MAIN STEAM PIPING SYSTEM	FIGURE 3.6-147a

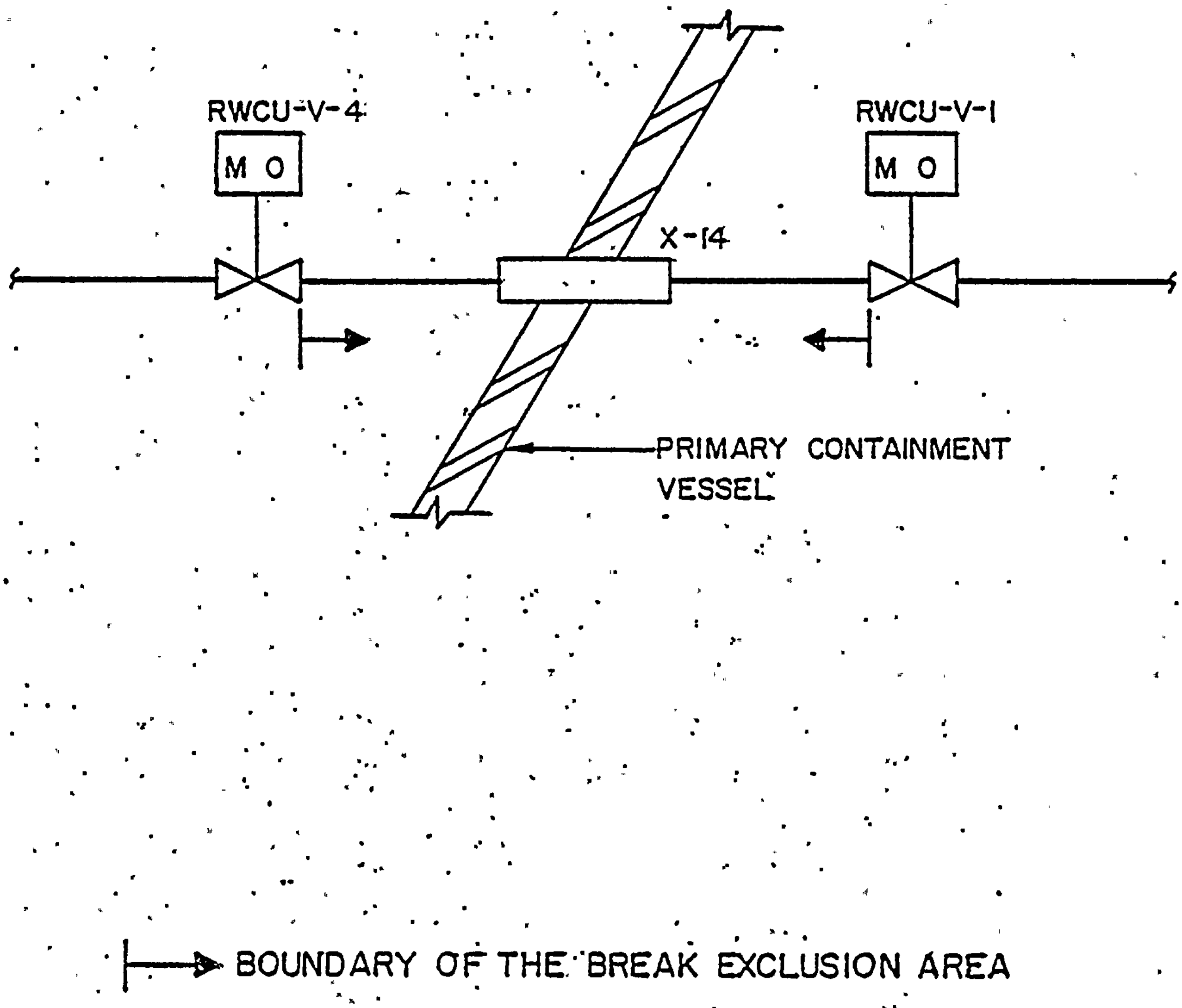


WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	REACTOR FEEDWATER PIPING SYSTEM	FIGURE 3.6-147b

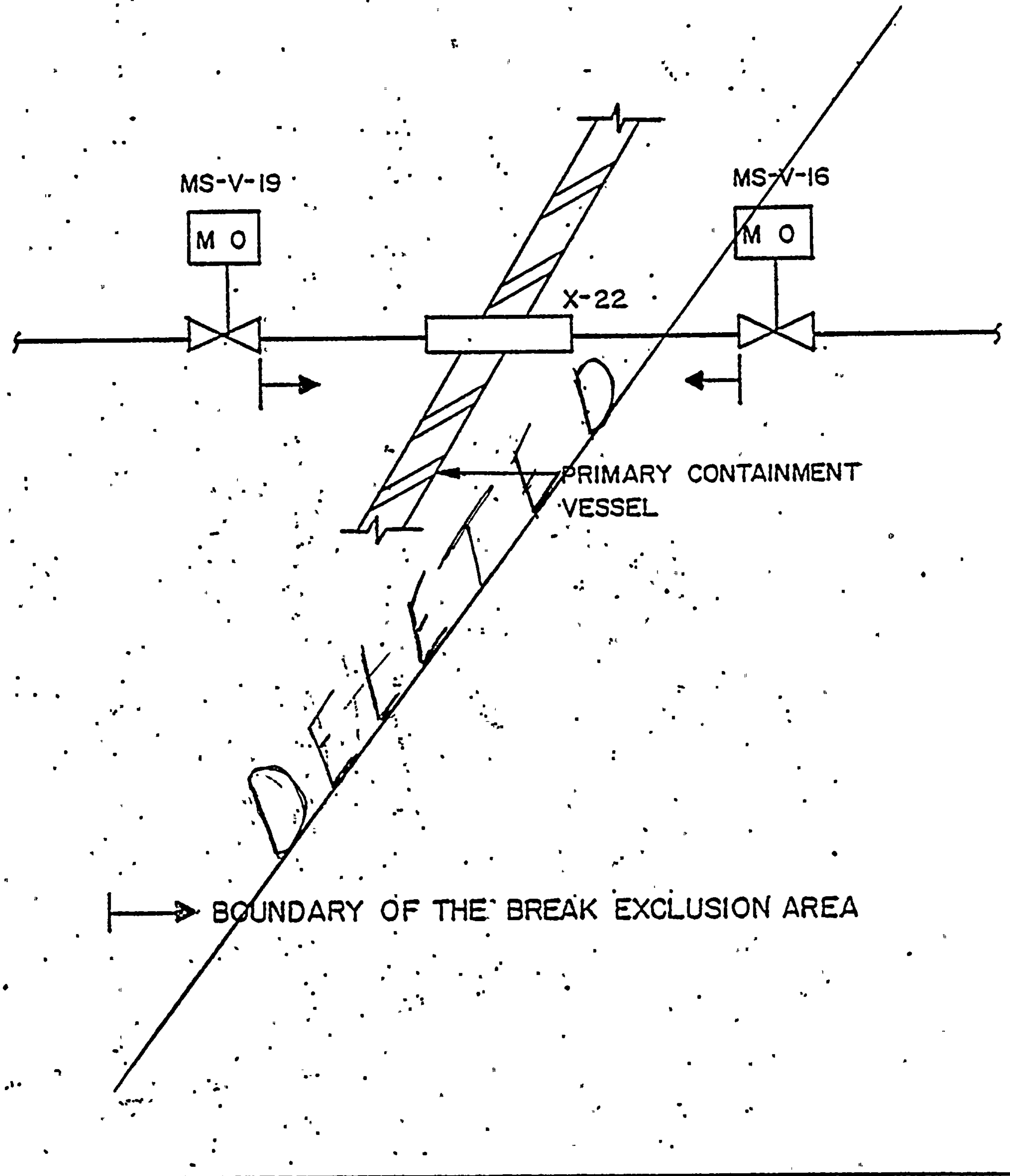


BOUNDARY OF THE BREAK EXCLUSION AREA

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	RHR CONDENSING MODE/RCIC TURBINE STEAM PIPING SYSTEM	FIGURE 3.6-147c



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	REACTOR WATER CLEAN UP PIPING SYSTEM	FIGURE 3.6-147d



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	MAIN STEAM VALVES DRAINAGE PIPING SYSTEM	FIGURE 3.6- 147e

QUESTION NO. 5
(3.6.2)

Any instances with limited break openings or break opening times exceeding one millisecond must be identified. Any analytical methods, representing test results or based on a mechanistic approach, used to justify the above must be provided and explained in detail. This applies to containment and annulus pressurization as well as general pipe break.

RESPONSE

In all analyses, except annulus pressurization analyses, full break with area equivalent to the pipe cross section is postulated to occur instantanesouly. No mechanistic approach is used.

In analyses related to annulus pressurization, the instantaneous approach is used in jet impingement and pipe whip restraint loads calculation. For the pressure time history, the recirculation line is postulated to break instantaneously producing full blowdown force. Subsequently, the broken end is assumed to separate in a finite time based on momentum and energy consideration. These analyses will be documented in detail as an appendix to the FSAR in the New Loads update.

Summation - This item is closed.

3.6.2 Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

Question 6

Expand paragraph 3.6.2.5.4.11c to provide assurance that sufficient protection has been provided to preclude the pipe break damage for main steam and reactor feedwater piping inside the main steam tunnel.

Response:

Please refer to revised 3.6.2.5.4.11 for the information requested.

Summation - This item is closed.

c. Verification of Pipe Whip Protection Adequacy

Sufficient pipe whip protection is provided for the RPV head vent piping to assure safety as defined in 3.6.2.5.2. There are no safety related systems in the vicinity of the RPV head vent piping and pipe whip restraints are provided to protect the primary containment structure.

3.6.2.5.4.11 Main Steam and Reactor Feedwater Piping Inside Main Steam Tunnel

a. System Arrangement

The four, 26-inch main steam and two, 24-inch reactor feedwater lines inside the main steam tunnel originate at the primary containment penetrations and run horizontally to the end of the tunnel. At this point, the six lines drop vertically and are then routed horizontally within the turbine generator building. An isolation valve is located in each line just beyond the penetration.

b. Pipe Whip Protection

The postulated pipe breaks and pipe whip restraints for the main steam and reactor feedwater lines inside main steam tunnel, are shown in Figures 3.6-33a and 3.6-34a. Where breaks are postulated, the six lines are restrained to prevent unacceptable motion. The restraints are mounted on steel structures which then tie into the concrete walls and floors.

c. Verification of Pipe Whip Protection Adequacy

Sufficient pipe whip protection is provided for the main steam and reactor feedwater lines inside the main steam tunnel to assure safety. ~~as defined in 3.6.2.5.2.~~

Insert

~~The basis for providing protection in this area is to prevent pipe whip impact with adjacent isolation valves and to prevent pipe break damage escalation. The six lines and the six isolation valves in this area are located in close proximity to each other. A pipe break in one of~~

add attachment response

~~the six lines, if unrestrained, may result in pipe whip impact with adjacent isolation valves, possibly rendering them inoperative. Futhermore, unrestrained motion may cause impact with other lines, which may result in escalation of pipe breaks. Such a condition may unacceptably increase the severity of the initial pipe break.~~

3.6.2.5.4.12 Residual Heat Removal System (RHR) - Low Pressure Core Injection

a. System Arrangement

The RHR/LPCI piping consists of three, 14-inch loops whose arrangement is the same for two loops with the third loop being the mirror image of the other two. The piping originates at the reactor vessel at elevation 552 ft., rises vertically to elevation 563 ft. where there is a horizontal section with a check valve. This valve is normally closed, limiting the high energy portion of each loop. After the valve, the normally unpressurized section of piping drops to an elevation just below the main steam relief valve platform where it is routed to a penetration through primary containment at elevation 534 ft.

b. Pipe Whip Protection

The postulated pipe breaks and pipe whip restraints for the three RHR/LPCI mode piping loops are shown in Figures 3.6-20a, 3.6-21a and 3.6-22a. Where pipe breaks are postulated, the three piping loops are restrained to prevent unacceptable motion. The restraints for this system are mounted onto the sacrificial shield wall and also on structures which tie back to the sacrificial shield wall.

c. Verification of Pipe Whip Protection Adequacy

Sufficient pipe whip protection is provided for the RHR/LPCI mode piping to assure safety as defined in 3.6.2.5.2. The pipe whip restraints

Insert to Page 3.6-70:

Pipe whip restraints are provided to prevent pipe whip impact with the main steam or feedwater isolation valves. In addition, impact with adjacent main steam or feedwater lines is prevented. Refer to Figures 3.6-6g through 3.6-6k.

QUESTION NO. 7

Provide justification for utilizing the load factor of 1.15 for the equivalent static load method. The acceptance criteria of SRP 3.7.2 for the equivalent static load method is to apply a load factor of 1.5.

RESPONSE

a. Paragraph 3.7.2.1.8.2 has been revised to clarify the alternate simplified method of analyses. (See attached revision).

b. A summary from the study performed to verify the adequacy of the alternate simplified method is attached.

Summation - This item is closed.

An alternate simplified method of dynamic analysis is used for cold and/or limber piping systems. This is the Equivalent Static Load Method for piping. This method consists of applying constant horizontal and vertical load factors conservatively derived from seismic floor response spectra.

The description of the method is as follows: Enveloped seismic building response spectra are derived from widened seismic floor response spectra. (The widening of the building response spectra is described in 3.7.2.5).

The piping is supported seismically such that the ~~piping fundamental frequency is higher than the building fundamental frequency, i.e., the piping system is more rigid than the building~~ first fundamental mode of all piping systems is above the spectral peak of the seismic response spectrum.

The piping systems are then represented by simply analytical models, e.g., simply supported beams. Initial maximum seismic support spans are analytically determined from the above model for the piping fundamental frequency. These maximum spans are modified, if required, so as not to exceed a conservative value of maximum stress based on ASME Code allowables, and a limiting piping deflection between supports.

insert attached

~~The building accelerations at the piping frequency are determined from the enveloped seismic building response spectra. These accelerations are increased by a minimum factor of 1.15 to include the effect of higher modes of vibration. These increased accelerations (g-values) are the load factors.~~

~~The horizontal and vertical loading factors are combined in the same way as described above for the detailed dynamic analysis.~~

3.7.2.1.8.3 Dynamic Analysis of Equipment

Equipment is idealized by a mathematical model consisting of lumped masses connected by elastic members or springs. Results for selected Category I equipment are given in Table 3.9-2. The dynamic response of the system is calculated by using the response spectrum method of analysis. When the equipment is supported at two or more points at different elevations, the response specrum analysis is performed by using the response spectra at the elevation near the center of gravity of the equipment as the design spectra for the NSSS equipment, and for balance of plant using the envelope of response spectra for supports. Modal maxima are combined as described in 3.7.2.1.5. The analyses are performed assuming the horizontal ground motion to act in either of two orthogonal directions, North-South and East-West. Maximum stresses

resulting from any one horizontal or vertical excitation are considered to act simultaneously and the absolute values are added directly, as described in 3.7.2.6 and 3.7.2.7.

The relative displacements between anchors are determined from the dynamic analysis of the structures. All cases of relative displacement between anchors are considered. If significant, these relative displacements are then used in a static analysis to determine additional stresses imposed on equipment. Further details are given in 3.7.2.1.8.3.1 for the NSSS equipment and 3.7.3.9 for all other equipment. The cases where the relative displacements between anchors are insignificant and

Insert to Page 3.7-15

In the application of the alternate simplified method on WNP-2, a conservative static "g" loading was chosen for all piping systems when this approach was used irrespective of the building or building elevation. This simplifies the work and results in different amounts of conservatism for different piping systems. To confirm the adequacy of the alternate simplified method, a study is performed for several representative piping systems. Pipe stress and pipe support loads are calculated for these representative systems using response spectrum analysis methods. Results are examined to confirm that pipe stresses are within allowables and pipe support loads are less than these calculated using the simplified method.

Justification

A study was conducted to demonstrate that the equivalent static analysis criteria employed on WNP-2 is conservative as compared to response spectrum analysis methods.

A. Approach
Dynamic response spectra analysis was performed on the sample problems. The results from the equivalent static method and response spectrum method were compared.

B. Piping Systems Studied *
1. Six (6) Seismic Category I piping systems were selected as representative systems.

2. The piping systems chosen represent a variety of sizes from 3" to 24" pipe diameters.

3. Two systems were chosen from each of the Seismic Category I structures; the Reactor Building, the Service Water Pump House and the Diesel Generator Building.

C. Results of the Study
The results of the equivalent static analysis and the response spectrum analysis were compared for each piping system studied and is summarized as follows:

1. Where piping system design was established by the equivalent static analysis criteria, analyses using the response spectrum method have shown that stresses are well within code allowables in all cases.

2. A total of 192 pipe supports were compared. A summary of Support Loads for 30 of the 192 supports is shown in Table 1 and lists in ascending order the load ratio for the five (5) supports with the smallest load ratio in each system. The minimum ratio (load ratio) of equivalent static method/response spectrum method shown in Table 1 is 1.80.

D. Conclusion
The Equivalent Static Analysis Criteria used in piping analysis on the WNP-2 project is conservative and provides an adequate basis for piping system design.

* Additional considerations factored into the study to ensure that the six systems chosen for study represent a conservative basis for comparison:

(1) Piping systems located at higher elevations in the building were chosen for study, since seismic response spectra at higher elevations are larger, in order to represent a conservative basis for comparison.

(2) Piping systems which are extensive in length and thereby exhibiting a variety of configurations and spans were chosen.

TABLE 1 - RESULT OF STUDY   Attachment 1

System	Location	Support Mark #	Calculated Load (lb) Equiv.** Static Method	Calculated Load (lb) Response Spectrum Method	Equiv. Static Method/ Response Spectrum Method
Diesel Generator Air Intake	Diesel Generator Bldg.	DE-7 DE-7 DE-8 DE-14 DE-14	4696 4860 3120 4124 10505	2347 2281 1411 1780 2960	2.0 2.1 2.2 2.3 3.5
Loop B Return DG ENG 1B DMA-CC-21	Diesel Generator Bldg.	SW-261 SW-263 SW-253 SW-258 SW-257	854 535 1085 1128 873	173 107 151 145 112	4.9 5.0 7.2 7.8 7.8
Standby Service Water Pump-house Spray Pond Cross-over	Service Bldg.	SW-10 SW-11 SW-186 SW-187 SW-181	717 926 1109 1109 961	268 263 277 262 224	2.7 3.5 4.0 4.2 4.3
Standby Service Water Pump-house Spray Pond Cross-over	Service Bldg.	SW-1 SW-16 SW-17 SW-3 SW-14	579 563 1335 859 1429	330 286 527 326 445	1.80 2.0 2.5 2.6 3.2
Service Water from 20" SW Loop A	Reactor Bldg.	SW-322 SW-344 SW-324 SW-343 SW-321	359 253 683 426 490	178 67 147 83 87	2.0 3.8 4.7 5.1 5.6

** Design Basis

TABLE 1 - RESULT OF STUDY (continued)

System	Location	Support Mark #	Calculated Load (lb) Equiv.** Static Method	Calculated Load (lb) Response Spectrum Method	Equiv. Static Method/ Response Spectrum Method
Service Water from 20" SW Loop B	Reactor Bldg.	SW-382	268	79	3.4
		SW-384	105	30	3.5
		SW-377	211	56	3.8
		SW-383	243	57	4.3
		SW-376	496	77	6.4

** Design Basis

QUESTION NO. 8
(3.7.3)

Provide clarification of the statement in Paragraph 3.7.3.2.1 of the FSAR, "Based on Reference 3.7-10, which summarized data related to seismic histories presented in PSARs for many plants, it is conservatively assumed that combined effects due to seismic events of an intensity less than or equal to OBE intensity may be considered equivalent to two earthquakes of OBE intensity. Therefore, the lifetime number of earthquake cycle may range from 200 to 600 assuming 30 seconds of strong motion earthquake acceleration for each seismic event".

RESPONSE

See revised 3.7.3.2.1 of FSAR.

Summation - This item is closed.

K_j = Stiffness contribution of element j

W_i = Circular natural frequency of mode i

3.7.3 SEISMIC SUBSYSTEM ANALYSIS

The general approach to the seismic subsystem analysis is identical to those procedures described in 3.7.2 for seismic system analysis, except for the soil/structure interaction effects.

3.7.3.1 Seismic Analysis Methods

The seismic analysis method used to analyze Seismic Category I subsystems is described in 3.7.2.1.

3.7.3.2 Determination of Number of Earthquake Cycles

3.7.3.2.1 Number of Cycles for All Items Except NSSS Systems and Components

Assuming the mathematical model of strong motion earthquake acceleration described in 3.7.1.2 (T=15 seconds), the number of peaks and troughs, N, of the random process representing the structural response may be estimated. (Reference 3.7-3) The response of nuclear plant structures is controlled mostly by one governing mode, for the range of building frequencies normally encountered in nuclear plant facilities, (1.5 Hz to 6.0 Hz,) the number N is evaluated to be from 50 to 150. For a strong motion earthquake acceleration of 30 seconds in duration, N is from 100 to 300 for each seismic event. Fatigue evaluation due to a safe shutdown earthquake is not required by ASME Code, Section III since it qualifies as a faulted condition.

The operating basis earthquake is an upset condition and therefore must be included in fatigue evaluations according to ASME Code, Section III. The probability for the occurrence of a seismic event or OBE intensity is extremely low. Lower intensity earthquakes have a higher probability of occurrence. Based on Reference 10, which summarized data related to seismic histories presented in PSARs for many plants, it is conservatively assumed that combined effects due to seismic events of an intensity less than or equal to OBE intensity may be considered eqivalent to two earthquakes of OBE

As a minimum, 50 maximum stress cycles due to OBE are used for fatigue evaluations of BOP components.

~~intensity. Therefore, the lifetime number of earthquake cycles may range from 200 to 600 assuming 30 seconds of strong motion earthquake acceleration for each seismic event.~~

~~During an actual seismic disturbance, only a small percentage of these cycles occur at the maximum, or even at a significant stress level. Reference 10 states that 99.5% of the stress reversals occur below 75% of the maximum stress level, and 95% of the reversals lie below 50% of the maximum stress level (See Figure 3.7-26). Based on this data, it is assumed that a total lifetime number of maximum seismic load cycles of 60 is a conservative estimate of the number of cycles which will have a significant contribution to fatigue usage.~~

3.7.3.2.2 Number of Cycles for NSSS Systems and Components

To evaluate the number of cycles which exist within a given earthquake, a typical boiling water reactor building-reactor dynamic model was excited by three different recorded time histories: (a) May 18, 1940, El Centro NS component 29.4 sec; (b) 1952, Taft N 69° W component, 30 sec; and (c) March 1957, Golden Gate S80E component, 13.2 sec. The modal response was truncated such that the response of three different frequency bandwidths could be studied, (0^+-10 Hz, 10-20 Hz, and 20-50 Hz). This was done to provide a good approximation to the cyclic behavior expected from structures with different frequency content.

Enveloping the results from the three earthquakes and averaging the results from several different points of the dynamic model, the cyclic behavior, as given in Table 3.7-18, was formed.

Independent of earthquake or component frequency, 99.5% of the stress reversals occur below 75% of the maximum stress level, and 95% of the reversals lie below 50% of the maximum stress level. This relationship is graphically shown in Figure 3.7-26.

QUESTION NO. 9
(3.7.3)

Provide justification of utilizing one OBE intensity earthquake for design of the NSSS systems and components in Paragraph 3.7.3.2.2. Specifically, provide justification that the information in Paragraph 3.7.3.2.2. is applicable to the WNP-2 site.

RESPONSE

For the NSSS piping, 50 peak OBE cycles are used.

For other NSSS equipment and components, a generic study serves as the basis for 10 peak OBE cycles. As shown in the letter, R. Artigas to R. Bosnak, "Number of OBE Fatigue Cycles in the BWR NSSS Design", September 17, 1981, 10 peak OBE cycles can envelope the cumulative fatigue damage of hundreds of less severe earthquake cycles.

Accordingly, the FSAR is revised as attached.

Summation -
The applicant is to provide the comparison of the response spectra mentioned in the letter by November 20, 1981.
This item is closed.

9

intensity. Therefore, the lifetime number of earthquake cycles may range from 200 to 600 assuming 30 seconds of strong motion earthquake acceleration for each seismic event.

During an actual seismic disturbance, only a small percentage of these cycles occur at the maximum, or even at a significant stress level. Reference 3.7-10 states that 99.5% of the stress reversals occur below 75% of the maximum stress level; and 95% of the reversals lie below 50% of the maximum stress level (See Figure 3.7-26). Based on this data, it is assumed that a total lifetime number of maximum seismic load cycles of 60 is a conservative estimate of the number of cycles which will have a significant contribution to fatigue usage.

3.7.3.2.2 Number of Cycles for NSSS Systems and Components

3.7.3.2.2.2 Other NSSS equipment and components

To evaluate the number of cycles which exist within a given earthquake, a typical boiling water reactor building-reactor dynamic model was excited by three different recorded time histories: (a) May 18, 1940, El Centro NS component 29.4 sec; (b) 1952, Taft N 69° W component, 30 sec; and (c) March 1957, Golden Gate S80E component, 13.2 sec. The modal response was truncated such that the response of three different frequency bandwidths could be studied, (0^+-10 Hz, 10-20 Hz, and 20-50 Hz). This was done to provide a good approximation to the cyclic behavior expected from structures with different frequency content.

Enveloping the results from the three earthquakes and averaging the results from several different points of the dynamic model, the cyclic behavior, as given in Table 3.7-18, was formed.

Independent of earthquake or component frequency, 99.5% of the stress reversals occur below 75% of the maximum stress level, and 95% of the reversals lie below 50% of the maximum stress level. ~~This relationship is graphically shown in Figure 3.7-26.~~

3.7.3.2.2.1 NSSS Piping

Fifty peak OBE cycles are postulated for fatigue evaluation.

(9)

In summary, the cyclic behavior number of fatigue cycles of a component during an earthquake is found in the following manner:

a. The fundamental frequency and peak seismic loads are found by a standard seismic analysis.

b. The number of cycles which the component experiences are found from Table 3.7-18 according to the frequency range within which the fundamental frequency lies.

c. For fatigue evaluation, one-half percent (0.005) of these cycles are conservatively assumed to be at the peak load and 4.5% (0.045) are assumed to be at or above three-quarter peak. The remainder of the cycles has negligible contribution to fatigue usage.

The safe shutdown earthquake has the highest level of response. However, the encounter probability of an SSE is so small that it is not necessary to postulate more than one SSE during the 40 year plant life. Fatigue evaluation due to the SSE is not necessary since it is a faulted condition and thus not required by ASME Code Section III.

The OBE is an upset condition and, therefore, must be included in fatigue evaluations according to ASME Code Section III. An investigation of seismic histories for many plants shows that during a 40 year life, it is probable that five earthquakes with intensities one-tenth of the SSE intensity, and one earthquake approximately 20% of the proposed SSE intensity, will occur. ~~Therefore, the probability of even an OBE is extremely low.~~ To cover the combined effects of these earthquakes and the cumulative effects of even lesser earthquakes, ~~one OBE intensity earthquake is~~ ten peak OBE cycles are postulated for fatigue evaluation.

Table 3.7-19 shows the calculated number of fatigue cycles and the number of fatigue cycles used in design.

3.7.3.3 Procedure Used for Modeling

The procedure used for modeling for the subsystem dynamic analysis is described in 3.7.2.3.

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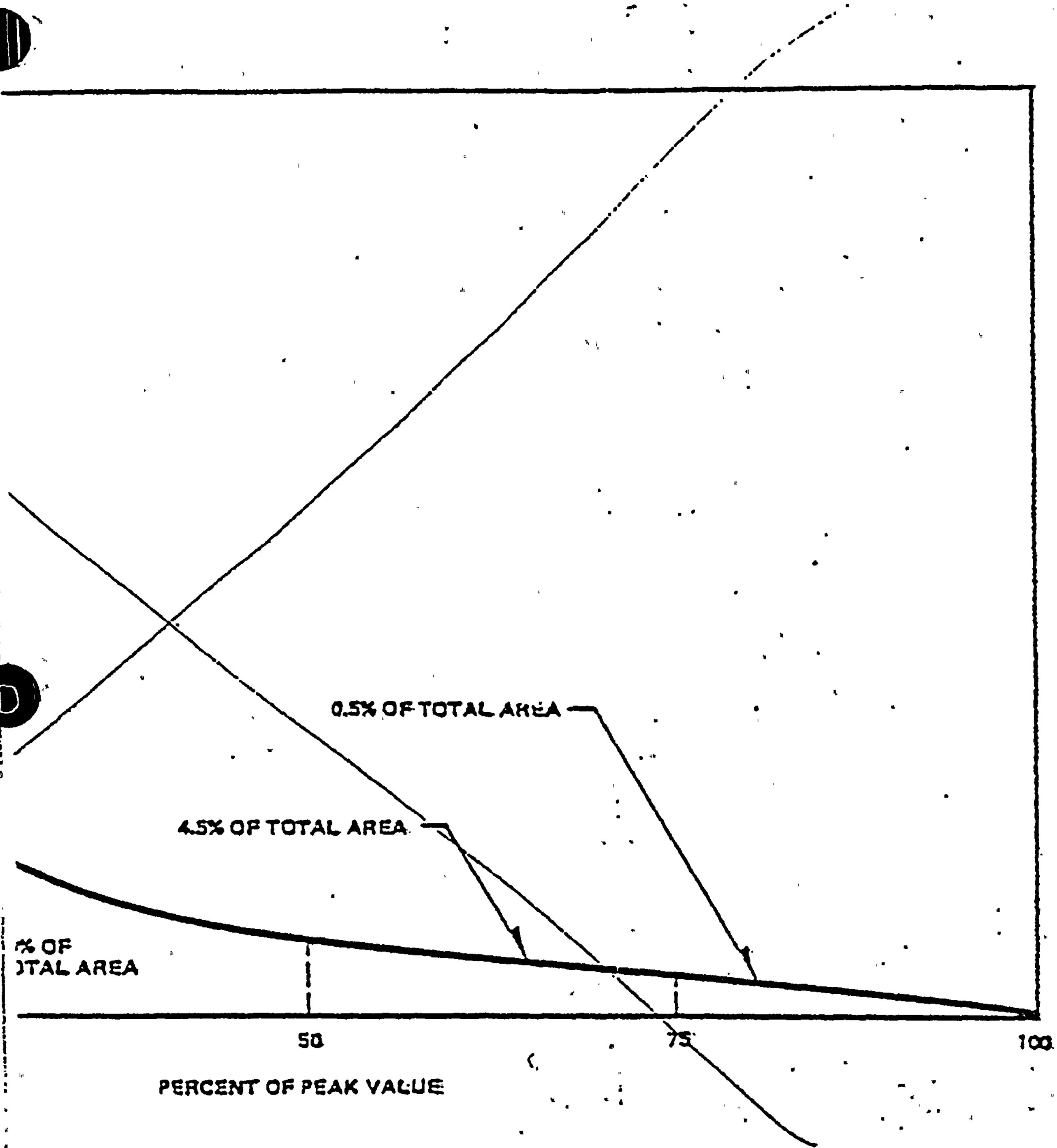
TABLE 3.7-18

NUMBER OF DYNAMIC RESPONSE CYCLES EXPECTED DURING A SEISMIC EVENT FOR NSSS SYSTEMS AND COMPONENTS

	Frequency Bandwidth (Hertz)		
~~Frequency Band (Hz)~~	0^+ - 10	10 - 20	20 - 50
Total Number of Seismic Cycles	168	359	643
Seismic Cycles at Peak Load	0.8	1.8	3.2
Seismic Cycles at or above 75% of Peak Load	7.5	16.2	28.9

No. of seismic cycles (4.5% of total) between 50% and 75% of peak loads

No. of seismic cycles (0.5% of total) between 75% and 100% of peak loads



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WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	~~DENSITY OF STRESS REVERSALS~~	FIGURE 3.7-26