LIC 9/15/80

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
METROPOLITAN EDISON COMPANY	)	Docket No. 50-289
	)	(Restart)
(Three Mile Island Nuclear	)	
Station, Unit No. 1)	)	

LICENSEE'S TESTIMONY OF

ROBERT W. KEATEN, GEORGE R. BRAULKE AND GEORGE J. BRAZILL

IN RESPONSE TO

UCS CONTENTION NO. 12, UCS CONTENTION NO. 14 AND

UCS CONTENTION NO. 3

(SAFETY CLASSIFICATION)

OUTLINE

The purposes and objectives of this testimony are to respond to UCS Contention 12, which asserts that safety related equipment at TMI-2 failed to perform its safety functions due to the environment created inside the containment by the accident and that the accident environment exceeded the expected design basis event (DBE) environment; to respond to UCS Contention 14, which asserts that the TMI-2 accident demonstrated that systems and components presently classified as non-safety related can have an adverse effect on the integrity of the core; and to respond to UCS Contention 3, which asserts that the pressurizer heaters and associated controls should be classified as important to safety and meet applicable safety-grade design criteria. The testimony shows that instrument failures related to the TMI-2 accident posed no threat to the health and safety of the public and that the accident environment did not exceed the DBE environment except for flooding.

Efforts taken and ongoing to assure the proper qualification and performance of TMI-1 electrical equipment are described. The general design approach with regard to differentiating between safety and non-safety equipment is discussed.

The fact that there were no unfavorable interactions of non-safety grade equipment which degraded the performance of safety systems during the TMI-2 accident is presented. It is also demonstrated that operation of the pressurizer heaters and associated controls are not important to safety.

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INTRODUCTION

This testimony, by Mr. Robert W. Keaten, GPU Manager of Systems Engineering, Mr. George R. Braulke, GPU Senior Electrical Engineer, and Mr. George J. Brazill, Principal Engineer, Babcock & Wilcox Company, is addressed to the following contentions:

UCS CONTENTION NO. 12

> The accident demonstrated that the severity of the environment in which equipment important to safety must operate was underestimated and that equipment previously deemed to be environmentally qualified failed. One example was the pressurizer level instruments. The environmental qualification of safety-related equipment at TMI is deficient in three respects: 1) the parameters of the relevant accident environment have not been identified; 2) the length of time the equipment must operate in the environment has been underestimated; and 3) the methods used to qualify the equipment are not adequate to give reasonable assurance that the equipment will remain operable. TMI-1 should not be permitted to resume operation until all safety-related equipment has been demonstrated to be qualified to operate as required by GDC 4. The criteria for determining qualification should be those set forth in Regulatory Guide 1.89 or equivalent.

The Board limited this contention to "equipment important to safety in the containment building and auxiliary building," and accepted it as limited (See Board First Special Prehearing Conference Order, dated December 18, 1979, at 21). UCS subsequently withdrew its sponsorship of UCS Contention No. 12, which has now been adopted as a Board Questi n (See Board

Memorandum and Order of Prehearing Conference of August 12-13, 1980, dated August 20, 1980, at 6.)

UCS CONTENTION NO. 14

The accident demonstrated that there are systems and components presently classified as non-safety-related which can have an adverse effect on the integrity of the core because they can directly or indirectly affect temperature, pressure, flow and/or reactivity. This issue is discussed at length in Section 3.2, "System Design Requirements," of NUREG-0578, the TMI-2 Lessons Learned Task Force Report (Short Term). The following quote from page 18 of the report describes the problem:

"There is another perspective on this question provided by the TMI-2 accident. At TMI-2, operational problems with the condensate purification system led to a loss of feedwater and initiated the sequence of events that eventually resulted in damage to the core. Several nonsafety systems were used at various times in the mitigation of the accident in ways not considered in the safety analysis; for example, long-term maintenance of core flow and cooling with the steam generators and the reactor coolant pumps. The present classification system does not adequately recognize either of these kinds of effects that nonsafety systems can have on the safety of the plant. Thus, requirements for nonsafety systems may be needed to reduce the frequency of occurrence of events that initiate or adversely affect transients and accidents, and other requirements may be needed to improve the current capability for use of nonsafety systems during transient or accident situations. In its work in this area, the Task Force will include a more realistic assessment of the interaction between operators and systems."

The Staff proposes to study the problem further. This is not a sufficient answer. All systems and components which can either cause or aggravate an accident or can be called upon to mitigate an

accident must be identified and classified as components important to safety and required to meet all safety-grade design criteria.

The Board limited this contention to the "core cooling system," and accepted it as limited (See Board First Special Prehearing Conference Order, dated December 18, 1979, at 23.)

UCS CONTENTION NO. 3

> The Staff recognizes that pressurizer heaters and associated controls are necessary to maintain natural circulation at hot stand-by conditions. Therefore, this equipment should be classified as "components important to safety" and required to meet all applicable safety-grade design criteria, including but not limited to diversity (GDC 22), seismic and environmental qualification (GDC 2 and 4), automatic initiation (GDC 20), separation and independence (GDC 3 and 22), quality assurance (GDC 1), adequate, reliable on-site power supplies (GDC 17) and the single failure criterion. The staff's proposal to connect these heaters to the present on-site emergency power supplies does not provide an equivalent or acceptable level of protection.

RESPONSE TO UCS CONTENTION NO. 12

BY WITNESS BRAULKE:

The contention implies that safety related electrical equipment failed to perform its safety functions at TMI-2 due to the environment inside containment created by the accident of March 28, 1979. This implication is not correct. It also implies that the environment exceeded the environment expected

for the design basis event (DBE). This is true only to the extent that certain safey-related electrical instrumentation was flooded following completion of its safety function. Further, independent of the TMI-2 accident, GPU is presently involved in a rigorous systematic review of the environmental qualification of safety-related electrical equipment at TMI-1. The results of this investigation are being submitted to the NRC for evaluation in response to their IE Bulletin 79-01B. The culmination of these efforts will provide assurance that TMI-1 safety-related equipment is adequately qualified for environmental conditions.

Table 1 lists the instruments at TMI-2 that failed during the period from March 28, 1979 through April 30, 1979. As seen in Table 1, all instruments functioned as designed, correctly indicating the process variables, without failure during the initial stages of the accident. No important instruments for monitoring the plant during the stabilization, cooldown and transition to natural circulation cooling were completely lost for this period of time. This allowed sufficient time to devise replacement methods of measurement, to install and test new systems, and to plan effective alternate methods for controlling the plant.

Table 2 lists the instruments at TMI-2 that failed during the period of May 1, 1979 to August 21, 1980. Because of existing redundancy or the availability of alternative actions, the consequences of instrument failure during this period posed

no threat to the health and safety of the public and did not reduce the ability to protect the health and safety of the public.

Although the exact cause of the failures cannot be determined until the instruments have been retrieved from the TMI-2 Containment Building and analyzed, engineering judgment suggests that the majority of failures are due to submergence in the reactor coolant in the containment building sump, a condition beyond the instrument design basis. The maximum expected flood level inside containment at TMI-1 has been calculated and any important instruments located below that level are being relocated. GPU is a member of the TMI-2 Instrument and Electrical Equipment Survivability Planning Group along with the Electric Power Research Institute, NRC and the Department of Energy. This group is concerned with understanding the effect on in-containment instrumentation and electrical equipment of prolonged exposure to various hostile environments and with understanding the adequacy of qualification testing.

The TMI-2 accident environment inside containment was less severe than the environment for a design basis event loss of coolant accident (LOCA). At TMI-2, the in-containment temperature was recorded at a peak of 190°F during the accident, was approximately 150°F for the first 14 hours, was less than 130°F through the fourth day of the accident, and was less than 100°F thereafter. The containment pressure reached a recorded peak

during the hydrogen burn of 28 psig, was approximately 1.5 psig for the first 14 hours, and was less than 0 psig thereafter. Studies by Babcock & Wilcox and Bechtel have estimated the one-year cumulative radiation dose to be approximately 3.6×10^6 Rads (gamma) and approximately 4.4×10^7 Rads (beta).(1,2,3) The in-containment humidity has been 100% for extended periods. Containment chemical spray (sodium hydroxide and boric acid) was automatically actuated for six minutes on March 28, 1979 in response to pressure generated by the hydrogen burn. Sump water level has been approximately eight to nine feet (approximately the 290 foot elevation) above the containment floor.(1) From Table 3 it can be seen that the accident environment experienced at TMI-2 was well below the levels determined for the DBE with the exception of flooding, and, as previously mentioned, appropriate measures have been applied to TMI-1 to avoid similar problems.

GPU has had and continues to have an interest and involvement in the subject of environmental qualification of electrical equipment. GPU has been a participating member of the EPRI/Utility Equipment Qualification Group since the inception of that group and continues its participation in the Babcock & Wilcox Owners Group Environmental Qualification Subcommittee. GPU is currently involved in a rigorous review of the environmental qualification of TMI-1 electrical safety equipment subjected to harsh environments. This review is following the requirements and guidelines of NRC IE Bulletin 79-01B.

IE Bulletin 79-01B is specifically intended to obtain additional information needed to evaluate the adequacy of environmental qualification of safety related electrical equipment in operating reactors. The Bulletin requires all operating reactor owners, including Licensee, to:

1. Provide a master list by system of all electrical equipment exposed to a harsh environment and required to function under postulated accident conditions.

2. Provide evidence of environmental qualification for each equipment item listed.

3. Provide service condition profiles (i.e., temperature, pressure, etc., as a function of time) and equipment operating time requirements for each equipment item listed.

4. Evaluate the qualification of the equipment items listed against the NRC "Guidelines for Evaluating Environmental Qualification of Class IE Electrical Equipment in Operating Reactors" and provide a plan to resolve any deficiencies.

5. Identify the maximum expected flood level inside primary containment resulting from postulated accidents.

For TMI-1 inside containment, the DBE environmental parameters are approximately the same as listed for TMI-2 in Table 3. The radiation value of 2×10^7 Rads (gamma) was calculated by the NRC for typical PWR's with dry type

containments and is based on the release of 100% of the noble gases, 50% of the halogens, and 1% of the remaining solids (developed from maximum full power operation of the core). The specific plant design makes this calculation applicable to TMI-1. An analysis was also done for TMI-1 to establish radiation dose levels in areas where fluids are recirculated from inside containment to accomplish long term core cooling following a LOCA. In addition, instrument relocation is being performed at TMI Unit 1 so that no important instruments would be submerged under maximum expected flooding.

The NRC staff has been ordered by the Commissioners (4) to:

A. Complete its review of environmental qualification and publish Safety Evaluation Reports by February 1, 1981.

B. Insure that safety related electrical equipment in all operating plants is qualified to the heretofore mentioned Guidelines or NUREG-0588 by June 30, 1982.

C. Make a technical judgment regarding continued operation where qualification documentation is questionable.

In such cases, written justification for continued operation must be provided to the NRC for evaluation of the safety consequence. This justification could include discussions of the safety implications of such factors as the failure mode of the equipment, the presence of backup components or systems, and the difference between the qualified environment and the postulated environment.

The information required for all items of Bulletin 79-01B is being submitted for TMI-1. These submittals do, however, contain some equipment item information where data from equipment suppliers has not yet been received. That data and any required evaluations will be subsequently forwarded to the NRC.

GPU is continuing its efforts to assure adequate environmental qualification of equipment with full recognition of current regulatory programs. Completion of this systematic process will provide reasonable assurance that equipment used to protect the public health and safety will be environmentally qualified for conditions more severe than the TMI-2 accident.

TABLE 1

TMI-2 Instrument Failures From Time of Accident Through April 30, 1979 (First 34 Days)

Day From Accident	Instrument Failure	Safety Related	Non-Safety
7	RC-LT-1 Pressurizer level indication		X
22	RC-LT-2 Pressurizer level indication		X
28	NI-2 Source range nuclear instrument indication*		X
30	SP-1A-LT1 Steam Generator A wide range level indication		X
31	RC-1-LT3 Pressurizer level indication		X

Note: No important instruments for monitoring the plant during the stabilization, cooldown, and transition to natural circulation were completely lost during this period, allowing sufficient time to devise replacement methods of measurement, to install and test new systems, and to plan the use of effective alternate methods of controlling the plant.

* NI-1 Source Range, a redundant instrument, remains functional to date.

TABLE 2

TMI-2 Instrument Failures From May 1, 1979 to August 21, 1980

Instrument Failure	Safety Related	Non-Safety	Function
AH-HI-5032, 5090 Reactor Building Humidity Indication		X	Indication
CF-LT-4 Core Flood Tank Level Indication		X	Indication
HP-R-209, 210, 214 Reactor Building Atmosphere Radiation Monitor		X	Indication
IC-R-1091, 1092 Letdown Cooler Radiation Monitor		X	Indication
Incore Thermocouples (4 of 52) - Incore Temperature Indication		X	Indication
Incore SPND's Incore Neutron Detectors		X	Indication
NI-4 Intermediate Range Nuclear Instrument		X	Indication
*RC-3A-PT1, 2 Narrow Range Reactor Coolant System Pressure-Loop A	X		Input to RPS
RC-3A-PT3, 4 Wide Range Reactor Coolant System Pressure-Loop A	X		Input to ESFAS
RC-3A-PT5 Low Range Reactor Coolant System Pressure-Loop A		X	Indication
*RC-3B-PT1, 2 Narrow Range Reactor Coolant System Pressure-Loop B	X		Input to RPS

RC-3B-PT3 Wide Range Reactor Coolant System Pressure-Loop B	X		Input to ESFAS
RC-14A-DPT1, 2, 3, 4 Reactor Coolant System Flow-Loop A	X		Input to RPS
RC-14B-DPT1, 2, 3, 4 Reactor Coolant System Flow-Loop B	X		Input to RPS
RC-22 PT1, 2, 3, 4, 5, 6, 7, 8 Reactor Coolant Pump Seals Pressure Indication		X	Indication
SP-1A-LT2, 3, 4, 5 Steam Generator A Level Indication		X	Indication
SP-1B-LT1, 2, 3, 4, 5 Steam Generator B Level Indication		X	Indication

*The true status of these transmitters is unknown since the RCS pressure is well below their range of 1700-2500 psig. However, the physical location of these transmitters indicates that they are submerged.

TABLE 3

TMI-2 ENVIRONMENTAL CONDITIONS INSIDE CONTAINMENT

Parameter	TMI-2 DBE Environment	TMI-2 March 28, 1979 Accident Environment
Temperature	286°F peak	190°F peak
Pressure	56.2 psig peak	28 psig peak
Radiation	2x107 Rads (gamma)	3.6x106 Rads (gamma)*
Humidity	100%	100%
Chemical Sprays	Yes (=24 hours)	Yes (=6 minutes)

*One year approximate total cumulative dose

RESPONSE TO UCS CONTENTION NO. 14

BY WITNESS KEATEN:

The contention quoted above does not correctly categorize the role of non-safety systems in the TMI-2 accident, nor does it properly consider the adequacy of the existing design approach for safety systems at TMI-1.

The general design approach to assure the safety of the public is to provide multiple levels of control or protection features for expected operational events, expected transient conditions, or severe equipment failures or natural phenomena. The equipment used to provide the greatest assurances of protection for the most severe plant accidents, or to assure safe shutdown despite severe natural phenomena, is designed and constructed to the highest standards. Systems designed to less stringent but still rigorous standards are used to control less severe transients and normal operations. The acceptability of the less stringent standard lies in the reduced consequence if these systems fail during a transient or normal operation, and the fact that the resulting event is less severe than (i.e., bounded by) the design bases events for the systems relied upon to protect the public. In the event that these normal control systems fail to perform their function, they are backed up by fully safety-grade equipment capable of mitigating the resulting event.

The TMI-2 accident did not demonstrate that the inherent design capabilities of safety systems were inadequate to protect against failures in non-safety systems. There were no failures of safety-grade equipment to perform their intended safety functions, nor were there unacceptable interactions of non-safety grade equipment with safety systems. Inadequate core cooling occurred because the operator had been trained to focus on the wrong set of parameters, and was thus led to reduce high pressure injection flow prematurely. Had full HPI flow been maintained until a subcooling margin was re-established, as is now clearly required by operating procedures and training programs, core damage would not have occurred.

The discussion of lessons learned from the TMI-2 accident in NUREG-0578 in the area of system design (Section 3.2, at 17) does not identify deficiencies in safety system design. The section recognizes that "...Non safety systems were assumed to be nonfunctional for mitigation of accidents..." except that non-safety control systems were evaluated to ensure that they did not reduce significantly the capability of safety systems. The TMI-2 accident in fact provided additional insight into the positive results that can be obtained if non-safety systems are available and utilized.

In conclusion, the TMI-2 accident did not demonstrate that non-safety systems require upgrading. The TMI-2 accident did highlight the need to upgrade the operators' understanding of safety system functions and emergency procedures. Those

improvements have been discussed in Licensee's testimony on Detection of Inadequate Core Cooling.

RESPONSE TO UCS CONTENTION NO. 3

BY WITNESS BRAZILL:

UCS Contention 3 states that the pressurizer heaters and associated controls are necessary to maintain natural circulation and, therefore, should be classified as important to safety and meet applicable safety-grade design criteria. Contrary to this contention, the pressurizer heaters are not necessary for natural circulation. In addition, while the preferred mode for certain situations, natural circulation is not essential for assuring adequate core cooling. Accordingly, the pressurizer heaters need not be safety-grade except to the extent that they form a part of the reactor coolant pressure boundary.

Licensee's testimony in response to UCS Contentions 1 and 2 discusses methods available for core cooling. As demonstrated in that testimony, natural circulation cooling, while the normal mode of cooling if forced circulation flow is not available, is not essential. Core cooling can be accomplished by the feed and bleed mode utilizing only safety-grade systems and components - i.e., the borated water storage tank, the High Pressure Injection System, the pressurizer safety valves, the containment and the Low Pressure Injection System.

Natural circulation cooling, in turn, can be accomplished by maintaining Reactor Coolant System pressure with one of three methods: the normal mode utilizing the pressurizer heaters; solid water operation with the Makeup and Letdown System; or solid water operation with the High Pressure Injection System. This latter method is functionally the same as feed and bleed operation, except that the equipment may be operated for pressure control rather than for core cooling per se.

Therefore, operation of the pressurizer heaters and associated controls is not essential to safety. Portions of the heaters do form a part of the reactor coolant pressure boundary, however, and thus conform to associated safety-grade requirements, including applicable requirements of General Design Criterion (GDC) 4 and GDC 2 of Appendix A to 10 CFR Part 50. The remaining GDC's listed in the contention are not applicable.

BY WITNESS BRAULKE:

Since use of the pressurizer heaters is the normal method of pressure control, Licensee is making modifications which permit certain heaters to be powered by the on-site emergency power supplies. The basis for and a description of the modifications are given in section 2.1.1.3.1 of the Restart Report, and are also described in Licensee's testimony on Connection of Pressurizer Heaters to Diesel in response to UCS Contention 4.

BY WITNESS BRAZILL:

In summary, operation of the pressurizer heaters and associated controls is not essential to maintaining natural circulation nor is natural circulation required to assure adequate core cooling. The heaters are safety-grade to the extent that they form a portion of the reactor coolant pressure boundary and conform to applicable requirements.

REFERENCES

(1) Bechtel Power Corp. letter dated December 4, 1979, R. L. Rider to G.E. Kulynych (B&W), Phase 3, Bechtel/B&W meeting follow-up TMI-2 Containment Recovery Engineering, Bechtel Job No. 13587, BLBW-001.

(2) Babcock & Wilcox, RCS Component Evaluation, Task 27, Contract 595-7100, Rev. 0, May 31, 1980.

(3) TMI-Unit 2 Planning Study for Containment Entry and Decontamination, Supplement 1, Bechtel Power Corp.

(4) U.S. Nuclear Regulatory Commission Memorandum and Order, CLI-80-21, dated May 27, 1980.

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B.S., Physics, Yale University, 1957. Post-Graduate and Professional Courses in Mathematics, Engineering and Business, UCLA, 1960-1972.

Experience:

Manager, Systems Engineering Department, GPU Service Corporation, April 1978 to present. Responsible for the development and application of specialized analytical skills in such areas as nuclear core reloads and fuel management; plant dynamic and safety analysis; system generating plant process computers; control and safety systems analysis, and analysis of plant operating performance for nuclear and fossil plants. Served as Deputy Director of Technical Support at Three Mile Island during the post-accident period.

Program Manager, Light Metal Fast Breeder Reactor Technology, Atomics International Division of Rockwell International, 1974 to 1978. Managed research and development programs performed for U.S. Department of Energy, including programs in reactor physics, safety and component development.

Manager of Systems Engineering, Light Metal Fast Breeder Reactor Program, Atomics International Division of Rockwell International, 1968 to 1974. Responsible for performance of safety analyses, development of safety criteria and development of instrumentation, control and safety systems design.

American Representative to the OECD Halden Reactor Project in Norway, 1965-1968. Participated in research on nuclear fuel performance, application of digital computers to nuclear reactors, and on development and application of in-core instrumentation.

Supervisor of Engineering, Sodium Reactor Experiment, Atomics International, Division of Rockwell International, 1962-1965. Responsibilities included analysis and measurement of the nuclear heat transfer and hydraulic parameters of the reactor core and process systems; specification and installation of nuclear and process instrumentation; design and installation of new control systems.

Senior Physicist, Sodium Reactor Experiment, Atomics International, Division of Rockwell International, 1959-1962. Performed measurements and analyses of the nuclear and thermal parameters of the reactor.

Experimental Physics Group, DuPont Savannah River Plant, 1957-1959. Performed measurements and calculations of the nuclear parameters of the reactor lattices.

Honors and Professional Affiliations:

Member of the Nuclear Power Plant Standards Steering Committee of the American Nuclear Society.

Member and past Chairman of the LMFBR Design Criteria (ANS-54) Standards Committee of the American Nuclear Society.

Registered Professional Engineer (Nuclear Engineering), California.

Publications:

"Analysis of TMI-2 Sequence of Events Operator Response," presented to a special session of the American Nuclear Society Conference, San Francisco, November 1979; and to Edison Electric Institute Conference, Cleveland, October 1979.

"The Role of Instrumentation in the TMI-2 Accident," presented at the American Nuclear Society Conference, June 1980.

Safety and Environmental Aspects of Liquid Metal Fast Breeder Reactors" 35th Annual American Power Conference, Chicago, Ill., May 1973.

"Safety Aspects of the Design of Heat Transfer Systems in LMFBR's" International Conference on Engineering of Fast Reactors for Safe and Reliable Operation, Karlsruhe, Germany, October 1972.

"Safety Criteria and Design for an FBR Demonstration Plant," ASME Nuclear Engineering Conference at Palo Alto, Calif., March 1971.

"Evaluation of Thermocouples for Detecting Fuel Assembly Blockage in LMFBR's," American Nuclear Society Annual Meeting, Los Angeles, California, June 1970.

"A Mathematical Model Describing the Static and Dynamic Instability of the SRE Core II," Reactor Kinetics and Control, AEC Symposium Series 2. (Also published as NAA-SR-8431.)

"Reactivity Calculations and Measurements at the SRE," ANS Topical Meeting: Nuclear Performance of Power-Reactor Cores, September 1963.

"Measurement of Dynamic Temperature Coefficients by Forced Oscillations in Coolant Flow," Trans-American Nuclear Society 5, No. 1, June 1962.

"Analysis of Power Ramp Measurements with an Analog Computer," Trans-American Nuclear Society 5, No. 1, June 1962.

"Reflected Reactor Kinetics," NAA-SR-7263.

Many other reports covering analytical and experimental work.

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"Qualification of Safety-Related Equipment for Nuclear Power Generating Stations," Drexel University/IEEE, 1978.
Applied Protective Relaying Course, Westinghouse Electric Corporation, 1969.

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Senior Electrical Engineer, GPU Service Corporation, 1978 to present. Cognizant engineer on environmental qualification of safety-related electrical equipment with responsibility for evaluating vendor plans and documentation pursuant to IEEE-323-1974 criteria. Participated in preparation and development of nuclear power plant electrical system design criteria, equipment rating calculations and equipment specifications.

Product Engineer, Instrument Division, Westinghouse Electric Corporation, 1973 to 1978. Participated in the environmental and seismic qualification program for Westinghouse protective relays for use in nuclear power stations. Additionally, served as Instructor for the Westinghouse Applied Protective Course.

Negotiations Engineer, Westinghouse Electric Corporation, 1969 to 1973. Provided technical information and product proposals on protective relays and relay systems.

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Standard Plant Integrator, Babcock & Wilcox Company, Nuclear Power Generation Division, 1974 to March, 1980. Responsible for technical coordination and review of the B&W standard NSS and associated standard Safety Analysis Report (B-SAR-205).

Project Manager, Babcock & Wilcox Company, Nuclear Power Generation Division, 1972 to 1974. Responsible for administration of the NSS contracts for the Washington Public Power Supply System, Units 1 and 4.

Associate Project Manager, Babcock & Wilcox Company, Nuclear Power Generation Division, 1970 to 1972. Responsible for assisting the Project Manager in administration of the NSS contract for Arkansas Nuclear One, Unit 1 project.

Project Engineer, Babcock & Wilcox Company, Nuclear Power Generation Division 1968 to 1970. Responsible for review and coordination of NSS engineering for the Arkansas Nuclear One, Unit 1 project.

Reactor Engineer (Nuclear Plant Engineer), U.S. Atomic Energy Commission, Idaho Operations Office, 1962 to 1968. Responsibilities included technical review and coordination of various design, construction, and startup activities associated with the Advanced Test Reactor and Loss of Fluid Test Reactor.

Engineer-In-Training, U.S. Atomic Energy Commission, Idaho Operations Office, 1960 to 1962. Training included National Reactor Testing Station orientation, ORSORT, and on-the-job training in fluid system design at Ebasco Services, Inc., New York City.