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

JUL 1 6 1981

MEMORANDUM FOR: Robert L. Tedesco, Assistant Director for Licensing, DL

Thomas H. Novak, Assistant Director for Operating Reactors, DL

FROM: James P. Knight, Assistant Director for Components & Structures Engineering, DE

SUBJECT: REPORTING OF UNSATISFACTORY EPRI/PWR TEST RESULTS FOR CONTROL COMPONENTS, INC. POWER OPERATED RELIEF VALVE AND DRESSER MODEL 31739A SAFETY VALVE

The attached memorandum from EPRI for the week of July 2, 1981 discusses the results of both steam and water tests performed at Wyle-Norco on the Control Components, Inc. PORV and the results of steam tests at the CE-Windsor facility on the Dresser 31739A Safety Valve. Note that this is not the same Dresser safety valve discussed in our June 16, 1981 memorandum. As described in the EPRI memorandum, each of these valves failed an EPRI "screening criterion" in one or more of these tests. For each "failure", the applicable criterion is as stated in the memorandum.

It is our understanding that the Licensees and Construction Permit Holders that utilize or plan to utilize one or both of these valves and the NSSS vendors have been notified of these test results and have the responsibility for assessing the safety significance of the observed valve behavior for their plants.

Our information from EPRI indicates that the Control Components, Inc. PORV is being used or will be used on the following plants:

McGuire 1 and 2
Catawba 1 and 2

The Dresser 31739A Safety Valve is being used or will be used on the following plants:

Calvert Cliffs 1 and 2
Palisades
Midland 1 and 2
Oconee 1, 2 and 3
Crystal River 3
TMI-1
Millstone 2

Although the specific safety significance of these test results is still being evaluated, this information may be relevant for licensing board notification.

James P. Knight, Assistant Director
for Components & Structures Engineering
Division of Engineering

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Memorandum

EPRI

July 2, 1981

TO: DISTRIBUTION

FROM: John J. Carey

SUBJECT: S/RV TEST ACTIVITIES

The EPRI/PWR Safety and Relief Valve Test Program testing activities for the period of June 29 through July 2 were as follows:

WYLE

On Tuesday, June 30, two tests were performed on the CCI valve utilizing air pressure to open and close the valve. The first was a 2750 psia steam test, the second a transition, steam to 650^o water, at 2515 psia. During both tests the valve opened and closed on demand.

On Wednesday, July 1, two tests were performed. The first was a 2750 psia steam test with increased preload utilizing spring force only for valve closure. During this test the valve opened on demand. Upon signaling the valve for closure the valve remained open for approximately 3 seconds prior to closure. Valve closure occurred at an inlet pressure of 2220 psia. For this test the EPRI screening criteria was not met (i.e., failure to close immediately on demand). The second test was a 2540 psia, water seal simulation test utilizing air pressure to open and close the valve. For this test the valve opened and closed on demand.

On Thursday, July 2, three additional tests were performed on the CCI valve. The first test was a 2750 psia, steam test utilizing spring force only for valve closure. The valve opened on demand. Upon signaling the valve for closure, the valve remained open for approximately 3 seconds prior to valve closure. Valve closure occurred at an inlet pressure of 2210 psia. The EPRI screening criteria was not met. The second and third tests were 2750 psia, increased valve preload, with and without air pressure for valve closure, respectively. For both tests the valve opened and closed on demand.

Upon re-evaluation of all the high pressure, preload tests performed on the CCI valve, it was noted that for the preload test performed on June 18, the valve opened on demand; however, upon signaling the valve for closure, the valve remained open for approximately 2 seconds, closing at a pressure of 2225 psia. The weekly activity report issued on June 19 did not note this delay in closure.

This completes the matrix of testing on the Control Components relief valve. The valve will be disassembled and inspected on Tuesday, July 7.

Current plans call for installation of the Masoneilan relief valve on Tuesday, July 7, with resumption of testing on Wednesday, July 8.

COMBUSTION ENGINEERING

On Saturday, June 27, two 2500 psig, high ramp rate, low back pressure, steam tests were performed on the Dresser safety valve (31739A). For these tests the ring settings were gradually changed to increase the initial lift of the valve. For the first test the valve opened within ± 3% of the valve design set pressure. A maximum stem position of 82% of rated lift was achieved at 6% of the valve design set pressure. Rated flow was achieved.* The valve closed at a pressure greater than 2250 psig.

For the second test performed on Saturday (with adjusted ring setting), the valve opened at a pressure of 2595 psig, which is greater than ± 3% of the valve design set pressure. A stem position of 100% rated lift was achieved at 6% of the valve design set pressure. Rated flow was achieved.* The valve closed at a pressure of 2185 psig, which is below the EPRI blowdown pressure of 2250 psig. The EPRI screening criteria were not met.

On Monday, June 29, a repeat test of Saturday's second test was performed. For this test the valve opened within ± 3% of the valve design set pressure. A stem position of 100% rated lift was achieved at 6% of the valve design set pressure. Rated flow was achieved.* The valve closed at a pressure of 2150 psig, which is below the EPRI blowdown pressure of 2250 psig. The EPRI screening criterion was not met.

On Tuesday, June 30, with the ring settings in the same position as the last two tests, a high ramp rate, high back pressure, steam test was performed. The valve opened within ± 3% of the valve design set pressure. A stem position of 100% rated lift was achieved at 6% of the valve design set pressure. Rated flow was achieved.* The valve closed at a pressure greater than 2250 psig. Peak back pressure was 855 psig.

On Wednesday, July 1, a repeat of Tuesday's test was performed with a slightly reduced back pressure setting. For this test the valve opened at a pressure within ± 3% of the valve design set pressure. A stem position of 100% rated lift was achieved at 6% of valve design set pressure. Rated flow was achieved.* The valve closed at a pressure of 2220 psig, which is below the EPRI blowdown pressure of 2250 psig. The EPRI screening criterion was not met. Peak back pressure achieved during this test was 605 psig.

On Wednesday, a second high ramp rate, high back pressure test was performed, with adjusted ring settings to further extend the time the valve remained in a full lift position. For this test the valve opened within ± 3% of the valve design set pressure. A stem position of 100% rated lift was achieved at 6% of the valve design set pressure. Rated flow was achieved.* The valve closed at a pressure of 2185 psig, which is below the EPRI blowdown pressure of 2250 psig. The EPRI screening criterion was not met. Peak back pressure achieved for this test was 655 psig.

*Based on preliminary venturi flow data.

On Thursday, July 2, a high ramp rate, low back pressure, steam test, with the same ring setting used for the previous test, was performed. The valve opened at a pressure with ± 3% of the valve design set pressure. A stem position of 100% rated lift was achieved at 6% of the valve design set pressure. Rated flow was achieved.* The valve closed at a pressure of 2070 psig, which is below the EPRI blowdown pressure of 2250 psig. The EPRI screening criterion was not met.

JJC/WJB/sm

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

AUG 6 1981

MEMORANDUM FOR: Robert L. Tedesco, Assistant Director for Licensing
Division of Licensing

Thomas H. Novak, Assistant Director for Operating Reactors
Division of Licensing

FROM: James P. Knight, Assistant Director for Components & Structures Engineering
Division of Engineering

SUBJECT: REPORTING OF UNSATISFACTORY EPRI/PWR TEST RESULTS FOR DRESSER MODEL 31739A AND CROSBY MODEL 3K6 SAFETY VALVE

The attached memoranda from EPRI for the weeks of July 6, July 13 and July 18 discuss the results of steam and water tests performed at Wyle-Norco on the Masoneilan and Copes Vulcan (17-4 PH plug and cage) PORV's. These PORV's passed the EPRI "screening criteria" for all of the tests.

Both Safety Valves, however failed a screening criteria in one or more tests as described in the memoranda. For each "failure", the applicable criterion is as stated in the memorandum. Note that all of these tests were performed on steam with a short inlet piping configuration. These two valves have not been tested as yet with the long loop seal inlet piping configuration that is normally used on Westinghouse plants.

It is our understanding that the Licensees and Construction Permit Holders that utilize or plan to utilize these safety valves and the NSSS vendor have been notified of these test results and have the responsibility for assessing the safety significance of the observed valve behavior for their plants.

Our information from EPRI indicates that the Dresser 31739A Safety Valve is being used or will be used on the following plants:

Calvert Cliffs 1 and 2
Palisades
Midland 1 and 2
Oconee 1, 2 and 3
Crystal River 3
TMI - 1
Millstone 2

The Crosby 3K6 is being used or will be used on the following plants:

St. Lucie 1 and 2
Fort Calhoun 1

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Although the specific safety significance of these test results is still being evaluated, this information may be relevant for Board notification.

If a decision is made to forward this information to one or more Boards, we recommend that a copy of the July 22, 1981 memorandum from Frank Cherny to Robert J. Bosnak be forwarded to the Board also. A copy of the memorandum was previously distributed to you. It contains the minutes of the July 17, 1981 meeting between the staff and EPRI and the PWR Owners Group at which the status of all of the Safety Valve and PORV testing in the EPRI program was discussed.

James P. Knight, Assistant Director
for Components & Structures Engineering
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W. Johnston
R. Bosnak
D. DiIanni

EPRI

July 24, 1981

TO: DISTRIBUTION

FROM: John J. Carey JJC

SUBJECT: S/RV TEST ACTIVITIES

The EPRI/PWR Safety and Relief Valve Test Program testing activities for the period of July 18-24 were as follows:

WYLE

During the period Saturday, July 18 through Friday, July 24, eight tests were performed on the Copes Vulcan relief valve utilizing the 17-4 P.H. Plug and Cage. The tests were performed under steam, preload, water, steam to water transition and water seal simulation conditions. During all tests the valve opened and closed on demand. The valve was disassembled and inspected and no damage was observed-that would affect future valve performance.

The next valve to be tested is the Copes Vulcan relief valve utilizing the 316 w/stellite plug.

COMBUSTION ENGINEERING

Six tests were performed on the Crosby 3K6 safety valve this week.

Three high ramprate, high backpressure, steam tests were performed on Tuesday.

The first test was performed with an adjusted ring setting to improve valve blowdown. The valve opened at a pressure within ± 3% of the valve design set pressure. A stem position greater than 100% of rated lift was achieved at a pressure of 6% above the valve design set pressure. During the test the valve started to oscillate (chatter), subsequently the test was terminated. Blowdown date was not obtained. Peak back pressure was 865 psia.

The second test was performed with an intermediate ring setting and a slightly reduced backpressure. All EPRI screening criteria were satisfied. Peak backpressure was 715 psia.

The third test was performed with a third ring setting adjustment. Due to instrumentation problems encountered during the test, valve stem position was not recorded. The test was repeated on Wednesday.

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For the repeat test the valve cycled three times. For the first cycle, all EPRI screening criteria were satisfied. Peak backpressure was 700 psia. For the second cycle, the valve opened at a pressure of 3.3% below the valve design set pressure which is below the EPRI opening pressure criterion. The valve closed at a pressure within 10% of the valve design set pressure. The data for the third cycle has not yet been reduced.

The second test performed on Wednesday, was a high ramp rate, low back pressure, steam test with the intermediate ring setting adjustment. This ring setting was the same as that used for the second test performed on Tuesday. This ring setting was selected for the remainder of the tests on the Crosby 3K6 safety valve. For this test the EPRI screening criteria were satisfied. Peak backpressure was 230 psia.

On Friday, a steam to water transition, low flow test was performed. For this test the valve cycled two times. For the first cycle the EPRI screening criterion were satisfied. For the second cycle the valve opened at a pressure of 4.7% below the valve design set pressure which is below the EPRI blowdown pressure criterion. The valve closed at a pressure within 10% of the valve design set pressure.

JCC/WJB/mw

<u>DISTRIBUTION</u>

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Memorandum

EPRI

July 17, 1981

TO: DISTRIBUTION

FROM: John J. Carey JJC.

SUBJECT: S/RV TEST ACTIVITIES

The EPRI/PWR Safety and Relief Valve Test Program testing activities for the period of July 13-17 were as follows:

WYLE

During the period from Friday, July 10 through Wednesday, July 16 testing was performed on the Masoneilan relief valve. The tests were performed under steam, preload, water, transition and water seal simulation conditions. One additional full pressure, 330°F water test and two repeat tests were also performed. The two repeat tests had a slightly increased air supply pressure to the air actuator to improve valve opening time. A total of eleven tests were performed. For all tests the valves opened and closed on demand. The valve was disassembled and inspected by the Masoneilan valve representative. No damage was observed that would affect future valve performance. The cage to body gasket had washed out during testing.

The Copes Vulcan relief valve utilizing the 17-4 ph plug and cage was installed today. Testing is scheduled to start tomorrow, July 18.

COMBUSTION ENGINEERING

During this week four tests were performed on the Crosby 3K6 safety valve. This valve has a design set pressure of 2500 psia. The first three tests were low ramp rate, short duration, high backpressure, steam tests. Due to computer and instrumentation problems encountered during the first two tests, all data was not recorded.

The third test was performed on Thursday, July 16. For this test, the valve opened at a pressure within ± 3% of the valve design set pressure. A maximum stem position of 98% of rated lift was achieved. Rated flow was achieved.* The valve closed at a pressure of 2245 psia, which is .2% below the EPRI blowdown pressure criterion of 2250 psia. Peak backpressure for this test was 690 psia.

On Friday, July 17 the fourth test on the Crosby 3K6 safety valve was performed. This test was a high ramp rate, high backpressure, steam test. The valve opened at a pressure within ± 3% of the valve design set pressure. A maximum stem position of 99% of rated lift was achieved at a pressure of 6% above the valve design set pressure. Rated flow was achieved.* The valve closed at a pressure of 2225 psia, which is 1% below the EPRI blowdown pressure criterion of 2250 psia. Peak backpressure for this test was 620 psia.

The next test on the Crosby 3K6 safety valve is scheduled for Monday, July 20.

Memorandum

EPRI

July 10, 1981

TO: DISTRIBUTION

FROM: John J. Carey John J. Carey

SUBJECT: S/RV TEST ACTIVITIES

The EPRI/PWR Safety and Relief Valve Test Program testing activities for the period of July 6-10 were as follows:

WYLE

Installation of the Masoneilan relief valve took place this week. Testing was delayed due to facility boiler problems encountered during system heatup. The problems have been resolved. Testing of the Masoneilan relief valve is scheduled to start today.

COMBUSTION ENGINEERING

On Monday, July 6, a high ramp rate, high back pressure, steam test was performed on the Dresser 31739A safety valve. For this test the upper ring controlling the effect of back pressure on valve performance was adjusted. The valve opened at a pressure within + 3% of the valve design set pressure. A maximum stem position of 111% of rated lift was achieved at 6% of the valve design set pressure. Rated flow was achieved.* The valve closed at a pressure 2245 psig, which is below the EPRI blowdown pressure of 2250 psig. The EPRI screening criterion was not met.

The Dresser 31739A safety valve was removed from the test stand on July 8. The Crosby 3K6 safety valve is presently being installed for testing.

JJC/WJB/sm

U.S. NUCLEAR REGULATORY COMMISSION
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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

JOINT AFFIDAVIT OF JOHN F. STOLZ AND DOMINIC C. DIIANNI

John F. Stolz and Dominic C. DiIanni state under oath as follows:

1. I, John F. Stolz am a Branch Chief assigned to Operating Reactors Branch #4, Division of Licensing, Office of Nuclear Reactor Regulation of the U. S. Nuclear Regulatory Commission. I am currently responsible for managing the branch activities that include the review associated with TMI-1. In addition, I review and approve all Safety Evaluations prepared by the staff members of the Branch. A copy of my professional qualifications is attached.

2. I, Dominic C. DiIanni am a Project Manager assigned to Operating Reactors Branch #3, Division of Licensing, Office of Nuclear Reactor Regulation of the U. S. Nuclear Regulatory Commission. I am currently responsible for managing all of the review activities and licensing actions associated with the Prairie Island Nuclear Generating Station, Units Nos. 1 and 2. At the time when activities associated with the reviews of unsatisfactory test results of safety valves for TMI-1, I was responsible for coordinating the reviews of all activities and the preparation of the staff Safety Evaluations for TMI-1 that were not related to the restart hearing matters. A copy of my professional qualifications is attached.

3. The "NRC Staff's Report on Board's Comments Regarding Board Notification of Unsatisfactory Test Results of Safety Valves" was prepared by us and is true and correct to the best of our knowledge and belief.

John F. Stolz
John F. Stolz

Dominic C. DiIanni
Dominic C. DiIanni

Subscribed and sworn to before me this 11th day of September 1981.

Marilyn Jollenster
Notary Public

My Commission Expires: July 1, 1982

JOHN F. STOLZ

PROFESSIONAL QUALIFICATIONS

OPERATING REACTORS BRANCH NO. 4

DIVISION OF LICENSING

I am the Branch Chief of the Operating Reactors Branch No. 4 of the Division of Licensing, U. S. Nuclear Regulatory Commission. This Branch is responsible for the overall safety and environmental project management for assigned licensed operating power reactors that includes the review of technical and procedural aspects of proposed amendments to operating licenses. Operating plants having Babcock and Wilcox reactor systems have been assigned to this Branch.

I accepted an appointment with the technical staff of the NRC Regulatory organization in 1969 and was assigned as Senior Project Manager for safety review of Quad-Cities Station Units 1 and 2 and Mendocino Power Plant Units 1 and 2. From April 1972 to April 1980, I have had branch supervisory responsibility for the project management of licensing reviews of BWR 4/5 and 6 plants, PWR plants using Westinghouse, B&W, and Combustion reactors, and standard designs from General Electric, Westinghouse, Stone and Webster and Fluor Pioneer for preliminary design approvals. During 1974, I also participated in the staff review of the Reactor Safety Study that was subsequently released as WASH-1400. From April 1980 to March 1981, I was Branch Chief of the Systems Interaction Branch responsible for the development of criteria and methods that can be used to identify and evaluate common cause type of failures that can lead to adverse systems interactions.

I graduated from the City College of New York in 1942 with a Bachelor of Science Degree in Civil Engineering, obtaining a Master Degree in Civil Engineering from the University of Southern California in 1966. I have also taken additional graduate level courses in nuclear engineering, structural engineering and mechanical engineering at the University of California and New York University.

My experience following my undergraduate degree, from 1942 to 1951, included military service in the Air Force, a member of the Civil Engineering staff at the City College of New York, and structural engineering and field construction with several consulting engineering and industrial firms. From 1951 to 1953, I was employed with the consulting firm of Devenco Inc., where I worked on the structural design and analysis of the first nuclear powered submarines, Sea Wolf and Nautilus. In 1953, I joined the Atomic Energy Department of North America Aviation which subsequently became Atomics International Division of North American Rockwell Corporation. My starting position of Research Engineering involved design and analysis of reactor core and system components related to a sodium-graphite reactor development program. I subsequently became supervisor of a unit responsible for the design of supporting facilities for all nuclear power prototype plants and nuclear research facilities. In 1958, I was assigned as Project Engineer for the design of the plant, fuel handling and support systems for the Hallam Nuclear Power Facility, a 75 MWe sodium-graphite reactor plant at Hallam, Nebraska. In 1959, I was assigned as Project Engineer to modify the Organic Mod ated Reactor Experiment at the National Reactor Test Station in Idaho, which involved redesign of the reactor core, pressure vessel, fuel handling, instrumentation and control and process systems. From 1962 to 1965, I held the position of Group Leader directing the work of four supervised units assigned to support the development, design and qualification of compact nuclear reactor systems (SNAP Program) specifically in the areas of testing facilities required to simulate nuclear spaceflight environment, stress analysis, and mechanical and electrical design on the SNAP systems. From 1965 to 1966, I supervised a process systems unit responsible for systems design and analysis supporting the company's development projects on sodium cooled reactors, organic-moderated heavy-water cooled reactor and desalinization systems. In 1966, I spent a year as Assistant Project Manager for the preliminary design and development of a 500 MWe sodium cooled fast-breeder reactor plant, specifically responsible for developing concepts, testing programs and budgetary plans for the overall plant and fuel handling. In 1967, I assumed a project management assignment with the Autonetics Division of North American Rockwell on system analysis studies in water and transportation systems, including management of the contract studies for the State of California on the systems analyses of operations and maintenance for the California State Water Project.

I am a member of the American Society of Civil Engineers, have been past Chairman of the Nuclear Structural and Materials Committee in the Structural Division of the Society, and am still an active member of that Committee and the Publications Committee. I am registered as a Civil Engineer in the State of California and a Professional Engineer in the State of New York.

DOMINIC C. DIIANNI

PROFESSIONAL QUALIFICATIONS

OPERATING REACTORS BRANCH #3

I am a Project Manager assigned to Operating Reactors Branch #3, Division of Licensing, Office of Nuclear Reactor Regulation of the U. S. Nuclear Regulatory Commission. I am responsible for managing all review activities and licensing actions associated with the Prairie Island Nuclear Generating Station Units 1 and 2. I recently was responsible for coordinating the reviews of all activities for Three Mile Island Nuclear Station, Unit 1 (TMI-1) not related to the restart hearing matters.

I received a B. S. degree in Chemical Engineering from the University of Pittsburgh in 1952. I have also taken extension courses from the University of Ottawa (Chalk River, Ontario, Canada) in 1957, in reactor physics and reactor engineering.

From 1953 to 1955 I served in the U. S. Military Service and was given a rating of scientific and professional personnel. While in the service I was assigned to a U. S. Army Petroleum Laboratory in Oakland California where petroleum products purchased by the armed forces were analyzed.

From April 1955 to May 1963, I worked as an engineer at Westinghouse Atomic Power Division Bettis Plant. During this period I was involved in reactor fuel development and design of primary reactor systems. In addition, I was assigned as a resident engineer at the reactor site in Chalk River, Ontario, Canada for two years with responsibilities for conducting reactor fuel irradiation programs.

system (HVAC) inside containment, (4) reducing steam generator sludge levels by increasing steam generator blow down capacity (i.e., contributes to a reduction in steam generator tube denting, etc.). Since backfit engineering for nuclear power plants covers a variety of tasks as Project Engineer it was essential that I have thorough knowledge and experience in the design and operation of numerous reactor auxiliary systems normally associated with nuclear power plants. Furthermore, in this position, I coordinated the efforts of instrumentation and controls, electrical and structural engineers.

From October 1975 to the present I worked in the Office of Nuclear Reactor Regulation of the U. S. Nuclear Regulatory Commission both in the Mechanical Engineering Branch and the Operating Reactors Branches, Division of Licensing. As a senior staff member in the Engineering Branch typical assignments were to evaluate problems concerned with water hammer developed in feedwater systems, pipe cracks in the PWR chemical volume control systems, decontamination and steam generator chemical cleaning. As a Project Manager assigned to TMI-1, I was responsible for coordinating the reviews of all activities and the preparation of the Staff Safety Evaluations that were not related to the restart hearing matters.

I am a registered Professional Engineer in the State of Illinois and a member of the American Nuclear Society.

In summary, my qualifications span 26 years in the nuclear field during which principles of chemical, mechanical and nuclear engineering have been applied. In addition, management skills have been developed during the course of handling various nuclear programs.

These programs included the inpile irradiation of defective fuel assemblies, development of primary water chemistry program, development of decontamination programs and material testing. In addition, I was responsible for primary loop system design that included equipment selection (i.e., pumps, valves, heat exchangers, process instrumentation, etc.). Field experience included initial start up of reactor systems at the ETR site in ARCO, Idaho 1959/1961.

From May 1963 to June 1973 I worked as a Nuclear Engineer and Project Manager at the NASA Lewis Research Center in the Advance Reactor Division. I had direct responsibilities in thermionic fuel development for nuclear space power systems. My assignments related to developing UO_2 a fuel material (clad in tungsten) for thermonic reactors as a space power source. This work involved managing contracts and supervising in house efforts in reactor design, fuel assembly fabrication and irradiation testing. I have published technical papers at NASA regarding these matters.

From June 1973 to October 1975 I worked as Project Mechanical Engineer at Fluor Corporation, Chicago, Illinois in the Nuclear Division. As a Project Mechanical Engineer, I supervised highly technical and scientific staff in performing backfit engineering on operating nuclear power plants (i.e., Kewaunee, Prairie Island, Quad Cities & Dresden). In order to perform these duties, specialized knowledge of applicable industry codes and standards was essential. Some of the specialized tasks under my supervision consisted of: (1) modifying pipe support systems to handle high energies developed by slug flow at the pressurizer safety valves discharge pipes, (2) upgrading the main steam isolation valves, (3) upgrading air condition and ventilation