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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 SECOND SUPPLEMENTAL

TESTIMONY OF ROBERT W. KEATEN

IN RESPONSE TO BOARD QUESTION NO. 6

(EMERGENCY FEEDWATER RELIABILITY)

OUTLINE

The purposes and objectives of this testimony are to supplement testimony previously filed by Licensee in response to Board Question 6 (Emergency Feedwater Reliability) by showing that there is no sound basis upon which to postulate that there will be an unusually high challenge rate to, or probability of failure of, the TMI-1 EFW system.

INTRODUCTION

In response to Board Question 6, Licensee initially filed and presented the following direct evidence:

1. Licensee's Testimony of Gary R. Capodanno, Louis C. Lanese and Joseph A. Torcivia in Response to Board Questions 6.a, 6.b, 6.c, 6.g, 6.h, 6.i, 6.j and 6.k, following Tr. 5642;

2. Licensee's Testimony of Robert C. Jones, Jr. in Response to Board Questions 6.e and 6.f, following Tr. 4588; and,

3. Licensee's Exhibit No. 15, "TMI-1 Emergency Feedwater System."

Part 6.d of the question was answered at page 12 of Licensee's Testimony of Robert W. Keaten and Robert C. Jones in Response to UCS Contention Nos. 1 and 2 (Natural and Forced Circulation), following Tr. 4588.

In response to a clarification of Board Question 6 provided at the hearing session of November 5, 1980 (Tr. 4812, 4813), Licensee has filed "Licensee's Supplemental Testimony of Robert W. Keaten, Joseph J. Colitz and Michael J. Ross in Response to Board Question No. 6 (Emergency Feedwater Reliability)," dated November 25, 1980.

Administrative Judge Jordan postulated, at the hearing session of November 20, 1980, that:

1. B&W plants are more sensitive because of the once through steam generator design and they experience an unusually high EFW challenge rate of three per year. Tr. 6150, 6175, 6179-6180.

2. Emergency feedwater systems, on an industry-wide basis, have experienced a failure rate of 1 in 25 per reactor-year -- which is so high that reliance on safety-grade criteria should be rejected. Tr. 6169, 6179-6180, 6182-6183.

3. Consequently, it should be demonstrated that the overall reliability of the TMI-1 decay heat removal systems is such that the probability of failure is less than 10^{-6} per year. Tr. 6184, 6186-6187.

This testimony, by Mr. Robert W. Keaten, GPU Manager of Systems Engineering, supplements previous Licensee testimony in response to Board Question 6 and addresses the postulates advanced by Administrative Judge Jordan on November 20, 1980.

SECOND SUPPLEMENTAL RESPONSE TO BOARD QUESTION NO. 6

BY WITNESS KEATEN:

EFW Challenge Rates

The primary difference between the B&W nuclear steam supply system ("NSSS") design and other PWR designs is the B&W once through steam generator ("OTSG"), which results in a more rapid effect (compared to the U-tube steam generator) on primary system performance from any large change in secondary

system inventory. In addition, the volume of water on the secondary side of a plant with the U-tube design is larger than the comparable inventory in a B&W plant. The close coupling of the primary and secondary systems in the B&W design, combined with the relatively small liquid volume in the secondary side, creates the characteristic of the OTSG referred to as "sensitivity" and "responsiveness," which has been considered in the safety analyses for TMI-1.

While the design and operating characteristics of the OTSG are important and must be understood, they have little bearing on the question of how often main feedwater is lost or the emergency feedwater ("EFW") system is challenged.

The design of the main and emergency feedwater systems are normally the responsibility of the architect/engineer ("A/E"), rather than the NSSS supplier. These designs vary widely, reflecting the different views of the A/E or owner, the NSSS supplier control system interface needs, and the type of main turbine-generator chosen. There are several significant issues that impact the number of loss of main feedwater transients reported, and the significance of partial losses of main feedwater. These in turn are important in considering the challenges to the emergency feedwater system.

The degree of redundancy built into the feedwater-condensate train and the normal controls of redundant components directly influence the number of transients. Systems

that have no redundancy in components will either trip or runback if any of the major components in the train are lost. TMI-1 has redundant components (three) in the condensate, condensate booster and heater drain pumps such that the standby pump should start if one of the two operating pumps fail, without a resulting trip or runback. If one of the two main feedwater pumps trip, the plant will runback or trip.

In the event of a trip, the residual heat of the reactor is normally removed via the steam generators. The method of feedwater makeup, post-trip, is also a function of the A/E, owner, NSSS supplier interfaces. Many (if not most) non-B&W NSSSs are designed such that on a turbine trip, reactor trip, or loss of main feedwater, the post-trip supply is always from the emergency feedwater system. This is because the main feedwater control system is incapable of low flow modulating control without wide swings in flow. Thus each trip, regardless of cause, results in an EFW demand. Additionally, these systems are required to operate for normal plant startups and shutdowns. The TMI-1 design is capable of providing the necessary low flow requirements during post-trip, startup and shutdown with the main feedwater system. Only in the event of a total loss of the main feedwater system is the emergency feedwater system required. Therefore, the number of actual demands on the TMI-1 system are substantially lower than for other designs.

NUREG-0560, "Staff Report on the Generic Assessment of Feedwater Transients in Pressurized Water Reactors Designed by the Babcock & Wilcox Company" (May 9, 1979), has been cited as support for the proposition that the challenge rate to EFW systems at B&W plants is approximately three per year. Tr. 5971. NUREG-0560 documents an NRC Staff study, conducted shortly after the accident at TMI-2, to assess the effect of feedwater transients on B&W reactors. NUREG-0560 at 1-1. While reviewing the significant feedwater transients that had occurred at B&W plants, the Staff also reviewed the operating experience at all PWR plants from March, 1973, to March, 1979. For that period, the Staff does report that there were 9 B&W plants that had 27 feedwater transients -- or 3 per year, per plant. NUREG-0560 at 3-1. This figure, however, does not correspond to the incident chronology presented in NUREG-0560; so that, without further information on the Staff's data base, we are unable to confirm the results.

Even assuming that the Staff's figures are correct, however, the Staff was not reporting an EFW challenge (or demand) rate of 3 per year at B&W plants. The Staff was reporting, instead, on the number of "feedwater transients." The Staff did not define the term "feedwater transients" in NUREG-0560, but knowledgeable Staff personnel subsequently have reported that the events reviewed in NUREG-0560 were the cases where forced plant shutdown resulted from a feedwater system

malfunction.(1) Thus, the feedwater transients reported in NUREG-0560 are not all incidents where emergency feedwater was called upon because of some failure in the main feedwater system. This is confirmed by the event descriptions provided in NUREG-0560, at 3-1 to 3-16.

The study documented in NUREG-0560 has been described by Staff personnel as "cursory in nature," designed to see if "a vast difference" in feedwater related malfunctions existed for the various vendors.(1) While the Staff found a somewhat larger number of transient events for B&W plants, it was not felt to be an appreciably higher frequency than for the other vendors. NUREG-0560 at 3-1. The Staff also expressed the thought that the greater number of feedwater transients may have been due to the generally younger age of B&W plants. NUREG-0560 at 3-1. The somewhat greater frequency of feedwater related transients was not by itself, however, considered by the Staff to be a safety concern.(1) In any case, the feedwater transient frequency of 3 per B&W plant per year, reported in NUREG-0560, does not represent the frequency of demands upon the EFW system at TMI-1, or at B&W plants generally.

At our request, Mr. Robert H. Koppe of S. M. Stoller Corporation examined OPEC-2 data maintained by his firm, and described in his earlier testimony(2), on events where operating reports clearly state that main feedwater was lost.

Mr. Koppe examined data for the calendar years 1979 and 1980, for all PWRs which were in operation prior to January 1, 1977, with the following exceptions: Indian Point 1, Yankee Rowe, San Onofre, Connecticut Yankee and TMI-1 (which was out of service). The sample consisted of 5 B&W units (10 unit-years of operation) and 23 Westinghouse or Combustion Engineering units (46 unit-years of operation). Mr. Koppe reports that the frequency of main feedwater losses that were reported at the B&W plants was 0.3 per plant per year, and that the frequency was 0.2 per plant per year for the Westinghouse/CE plants. There have been no instances of a loss of main feedwater during the operating history of TMI-1.

Consequently, I believe there is a sound basis upon which to postulate that the emergency feedwater system demand rate at TMI-1 (or at B&W plants generally) will be substantially less than 3 per year.

EFW Failure Rates

It has been postulated that the probability of failure of the TMI-1 EFW system is 1 in 25 per reactor-year. This is based upon testimony by NRC Staff witness Lantz, during cross-examination, that Licensee Event Reports indicate 8 failures of safety-grade EFW systems in 200 reactor-years. Tr. 6093, 6094. Mr. Lantz also testified that for all plants there were 9 EFW system failures in 280 reactor-years. Tr. 6106-6108.

There are several reasons why this data is not useful and cannot be tied to an EFW demand rate. First, Mr. Lantz was not reporting EFW system failures upon demand. He was providing routine LER data on system availability. This includes testing experience. In fact, NRC Staff witnesses have testified that data on EFW system success on demand is not available.(3) Consequently, the data tells us nothing about the probability that the TMI-1 EFW system will fail if it is called upon or demanded.

Second, the data is not applicable to TMI-1. Four of the eight failures reported by Mr. Lantz involved normal start-up operations. Tr. 6095, 6096. In contrast, the EFW system at TMI-1 is not normally used for plant startup or shutdown. The data also includes failures of associated systems which are not found at TMI-1. This illustrates the severe limitations on the application of industry-wide EFW system experience to a specific plant. Just as EFW system demand frequency is a function of plant specific factors, as I explained earlier, EFW system availability and operation upon demand is dependent upon plant specific EFW components, EFW support services, component and system testing, and maintenance.

For all of these reasons, I believe that there is a sound basis upon which to postulate that the emergency feedwater system failure rate at TMI-1 will be very much less than 1 in 25 per reactor-year.

Conclusion

Experience with the TMI-1 main and emergency feedwater systems has been excellent. There have been no failures of the EFW system on demand, nor have there been any total loss of main feedwater events (other than required tests) which would challenge the EFW system. Thus, the TMI-1 design has provided a stable, reliable main feedwater system that is capable of all normal operating and shutdown feedwater services, and it has in reserve a reliable EFW system fully capable of performing the necessary services under abnormal transient or accident conditions. I am aware of no reasonable basis upon which to project an unusually high demand rate for the TMI-1 EFW system or an unusually high failure rate of that system upon demand.

REFERENCES

1. NRC Staff Testimony of Mark P. Rubin and Thomas M. Novak Regarding the Acceptability of Feedwater Transients Referenced in NUREG-0560, following Tr. 1163 (March 6, 1980), In the Matter of Sacramento Municipal Utility District (Rancho Seco Nuclear Generating Station), NRC Docket No. 50-312.

2. Licensee's Testimony of Robert H. Koppe in Response to CLI-80-5, Issues 8 and 9 (Licensee's Infraction, LER and Operating Experience History), following Tr. 13,335.

3. NRC Staff Testimony of J. Wermeil, W. Jensen, E. Lantz, and B. Boger Regarding Emergency Feedwater System Reliability (Board Question 6), following Tr. 6035, at 3, 4.

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Education: 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.