UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

NRC STAFF TESTIMONY OF DONALD F. SULLIVAN
REGARDING DERIVATION OF INSTRUMENT INPUT SIGNALS

(ECNP Contention 1(c))

Q 1. Please state your name and position with the NRC.

A. My name is Donald F. Sullivan. I am a Senior Nuclear Engineer assigned to the Reactor Systems Standards Branch, Office of Standards Development. From July 1979 to September 1, 1980, I was temporarily assigned to the Instrumentation and Control Systems Branch, Office of Nuclear Reactor Regulation.

Q 2. Have you prepared a statement of professional qualifications?

A. Yes. A copy of this statement is attached to this testimony.

Q 3. Please state the nature of the responsibilities you have had with respect to the Three Mile Island, Unit 1 restart program.

A. I prepared the responses to UCS Interrogatories 67, 83, 85, 86, 88, 89, 90, 91, 92, 93, 94, 95, 97, and 98.

Q 4. What is the purpose of your testimony?

A. The purpose of my testimony is to respond to ECNP Contention 1(c), which states:

"The electronic signals sent to the control room in many cases record the wrong parameters, and may, thereby, mislead the reactor operator. For instance, in the case of the Electromatic Relief Valve ("ERV; the Metropolitan Edison designation is RC-RV2"), the signal sent to the control room to indicate a closure of this valve indicates only the electrical energizing of the solenoid which closes the valve, not the actual physical valve closing itself(4). This misleading signal aggravated the accident at TMI-2. There is no reasonable assurance that this same problem, or comparable ones, cannot arise many times over at TMI-1. It is the obligation of the Suspended Licensee to provide sufficient information on the performance capability of all pertinent components of the control system to reasonably ensure that electronic signals will record, accurately and in a timely manner, all necessary and correct parameters."

ECNP Contention 1(c), as modified by the Board, alleges that the Class 1E instrumentation in the TMI-1 control room by which the operator monitors information derived from signals from the containment isolation and core cooling systems following a feedwater transient and small break LOCA is inadequate in that it measures or records the wrong parameters and may mislead the operator.

ECNP Contention 1(d), as modified by the Board, alleges that the ranges of the Class 1E instruments referred to in ECNP Contention 1(c) are insufficient.

Because not all of the instruments used by the operator to monitor

information following a feedwater transient and a small break LOCA are Class 1E, the staff has chosen to:

(1) Identify in its partial response to ECNP Contention 1(d), sponsored by W. Jensen, et. al., the instruments used by the operator to perform necessary functions and monitor information derived from signals from the core cooling system and the containment isolation system following a feedwater transient or small break LOCA without regard to whether the instrumentation is labeled as Class 1E, and then to show that the ranges of the instruments are sufficient.

(2) Discuss in its reference to ECNP Contention 1(c) whether the signals to the instruments identified in the response to ECNP Contention 1(d) are derived from direct measures of the desired variables.

Q 5. Does the NRC have any regulation that requires the direct measurement of variables that are monitored by the operator following a feedwater transient and a small break LOCA?

A. Yes, but 10 CFR 50.55 a(h), IEEE 279-1968 and IEEE 279-1971, applies only to those monitoring instruments that are part of the protection system. Specifically, Section 4.8 of IEEE 279-1968 states, "To the extent feasible and practical, protection system inputs shall be derived from signals which are direct measures of the desired variables." (The 1971 version editorially substituted "that" for "which"). None of the instrument channels discussed in this response whose inputs are derived from signals which are not direct measures of the desired variables are part of the protection system.

Q 6. Are the designs of relief valves such the electromagnetic relief valve described in the contention subject to the provisions of IEEE-279-1968 or 1971?

A. I know of no instance where they are. Relief valves are considered to be control devices, rather than devices that are essential to safety, in terms of performing the pressure relief function. They serve to limit system pressure to values below the setpoints of the safety valves. The safety valves are the ultimate overpressure safety devices.

Since TMI-2, however, it has become apparent that more attention needs to be paid to the reclosure of both safety and relief valves in terms of more reliable position indication information available to the operator. The licensee is proposing to measure flow downstream of the electromagnetic relief valves and safety valves as a means of determining if the valves are open or closed. Although valve position, per se, is not monitored, the presence or absence of discharge flow from the valves bears a direct relationship to the "open" or "closed" state of the valves during reactor operation.

Q 7. Within the context of the contention, as limited by the Board and interpreted by the staff, are all other inputs to indications in the control room derived from signals that are based on direct measurement of variables?

A. My answer is based on the NRC Staff's partial response of Messers. W. Jensen, et. al., to Contention ECNP 1(d).

Inputs to certain of the instruments discussed in that testimony will be derived from signals that are direct measures of the desired variables, as follows:

"Steam Generator Level"

"Primary System Pressurizer Pressure"

"Core Exit Temperature"

"Hot Leg Temperature"

In addition, the position indications (open/closed) of the containment isolation valves discussed in the NRC staff testimony of Messrs. Jensen, et. al. are derived from directly actuated position limit switches.

Inputs to the following instruments will be derived from signals that are not, strictly speaking, direct measures of the desired variables. These instruments are not a part of the protection system since their signals do not actuate reactor trip or engineered safety features. Therefore, direct measurement of variables is not required by NRC regulations:

"PORV Position", and "Safety Valve Position": These variables are not measured by devices that directly sense valve position. "Open" or "Closed" is sensed by flow meters which monitor discharge flow when the valves are open and a no-flow condition when they are closed. Since an open valve will always result in a discharge flow, the staff believes that the measurement system constitutes a safe design. (See TMI Restart SER, NUREG-0680, Section 2.1.3.a, Pages C8-11, 12, 13.)

"Subcooling": There is no instrument that can directly measure the amount of subcooling (in the primary system) which is the number of degrees the liquid is below the boiling point for the system pressure at a given time. The system pressure and boiling point are, however, directly related by the laws of physics such that, for a given system pressure, there is a unique boiling point. Thus, a measurement of pressure also yields the boiling point. (See TMI Restart SER, NUREG-0680, Section 2.1.3.(b), Pages C8-16, 17, 18, 19).

Simply subtracting the measured temperature from the known boiling point yields the number of degrees that the liquid is subcooled.

In the TMI-1 system, temperature and pressure are measured directly. The inferred boiling point and subsequent computer calculation (Subtraction) result in an element of indirectness. We believe, however, that this measurement technique results in highly reliable "subcooling" information to the operator.

Q 8. Do you believe that this instrumentation by which the operator monitors information about the core cooling and containment isolation systems following a feedwater transient and a small break LOCA is adequate in that it measures or records appropriate parameters and thus will not mislead the operator?

A. Yes.

PROFESSIONAL QUALIFICATIONS

DONALD F. SULLIVAN

U. S. NUCLEAR REGULATORY COMMISSION

I am a senior nuclear engineer assigned to the Reactor Systems Standards Branch of the Office of Standards Development. I am currently on temporary assignment to the Instrumentation and Control Systems Branch, Office of Nuclear Reactor Regulation, performing various design reviews incident to the plant licensing process. This assignment will terminate on September 1, 1980.

I hold a Bachelor of Science degree in Physics from Holy Cross College, Worcester, Massachusetts, and a Master of Science degree in Physics from Trinity College, Hartford, Connecticut. In addition, I have studied electrical engineering, mathematics and physics at the graduate schools of Brown University, Providence, RI, and the University of Tennessee, Knoxville, TN.

I have approximately 26 years of professional experience, commencing in August 1954. During the first 9½ years, I was a member of the Instrument Group, and later the Controls Group, of the Connecticut Advanced Nuclear Engineering Laboratory (CANEL), Middletown, CT. This service included a temporary assignment of 19 months at the Reactor Controls Department, Oak Ridge National Laboratory. My responsibilities at CANEL included the design, specification and installation of various portions of the instrument and control systems for the CANEL critical assembly facilities, the Lithium Cooled Reactor Experiment and its simulator, and miscellaneous test stands.

I joined the AEC (NRC) in March 1964 and for the first (approximately) 8 years performed licensing safety reviews of the protection, control and emergency power systems of numerous commercial nuclear power stations and research and military reactors, and participated in the formulation of related standards and guides.

In April 1972, I was transferred to what is currently the Reactor Systems Standards Branch of the Office of Standards Development. In this capacity I am responsible for the development of various regulatory guides and criteria in the areas of protection, control, and emergency power system design and testing. From August 1978 to April 1979 I served as Acting Branch Chief of the Reactor Systems Standards Branch.

I am the NRC member of the IEEE Nuclear Power Engineering Committee, and participate in the Committee's development of standards for nuclear power plants.

In August 1973 I was the U. S. member of the International Atomic Energy Agency's Panel on the Code of Practice on Safe Reactor Design and Construction, held in Vienna, Austria.

I hold Patent No. 3,050,575 for the development of a special purpose thermocouple.

OUTLINE

This testimony of Donald F. Sullivan contains the NRC Staff's response to ECNP Contention 1(c).

The purpose of this testimony is to demonstrate that, contrary to the allegations made in the contention, the instrumentation in the TMI-1 control room by which the generator monitors information from the core cooling and containment isolation systems following a feedwater transient or small break LOCA receives signals based on measurements of appropriate parameters.

Conclusions to be drawn from this testimony:

-- NRC regulations that require direct measurement of variables that are monitored by the operator apply only to measurements that provide inputs to instruments that are part of the protection system.

-- None of the instrument channels by which the operator monitors information from the core cooling and containment isolation systems following a feedwater transient or small break LOCA, that receive inputs derived from signals that are not direct measures of the desired variables, are part of the protection system.

-- The instrumentation used to monitor steam generator level , pressurizer pressure, core exit and hot leg temperatures and containment isolation valve positions receive inputs derived from signals that are direct measures of the desired variables.

-- PORV and safety valve positions and core coolant subcooling are not directly but are indirectly measured.

-- The instrumentation by which the operator monitors information about the core cooling and containment isolation systems following a feedwater transient and small break LOCA measures or records appropriate parameters.

[redacted]