UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of HOUSTON LIGHTING AND POWER COMPANY (Allens Creek Nuclear Generating Station, Unit 1)	Docket No. 50-466

NRC STAFF TESTIMONY OF MARVIN W. (WAYNE) HODGES
REGARDING POSITION INDICATION FOR SRV

(DOHERTY CONTENTION 42)

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

A. My name is Marvin W. (Wayne) Hodges. I am employed by the U.S. Nuclear Regulatory Commisison as a Section Leader in the Reactor Systems Branch of the Division of Systems Integration. A copy of my professional qualifications is attached.

Q. What is the purpose of your testimony?

A. The purpose of this testimony is to respond to Doherty Contention 42, which reads as follows:

> Intervenor contends his health and safety interest will be injured because the information system giving the position of power operated relief valves and safety valves to the reactor operators is ambiguous and in need of improvement. This is a finding of the "Lessons Learned Task Force" on Three Mile Island, Unit 2 (TMI-2), reported in NUREG-0578, on p. 7, and is further supported by the fact Applicant states it ". . . will comply with the recommendations to alleviate this problem (Letter from E. A. Turner, Applicant to H. Denton, NRC, August 9, 1979, Attachment A, p. 2). However, Applicant does not say how this will be done, nor is there evidence this is even possible. Hence, this intervenor contends Applicant must show how the recommendation will be complied with at the construction license hearing.

Q. Is the position indication for safety/relief valves at ACNGS ambiguous and in need of improvement?

A. No. In Amendment 57 to the Allens Creek PSAR, the applicant has committed to a system in which safety/relief valve position indication will be determined by pressure sensors in the discharge pipe which are redundant, safety grade, seismically and environmentally qualified, and powered from Class 1E power sources. An alarm indicating that a safety/relief valve is open will be provided in the control room. This system will provide an unambiguous indication of valve position.

Q. How does this valve position indication differ from that proposed for Allens Creek and used in other reactors prior to the accident at Three Mile Island?

A. Prior to that accident, most, if not all, reactors used thermocouples to indicate flow through relief and safety valves. At TMI-2, both thermocouples and an indication of an open or closed signal to the power operated relief valves were used. Neither of these two signals is unambiguous. The open/close signal indicates what the valve was commanded to do, not what it did. The thermocouple can heat up due to a leaking valve or remain hot for a period of time after a valve has closed, thus providing the operator with a false position indication.

As a result of the lessons learned from TMI-2, all reactors are now required to have a direct, unambiguous indication of valve position for safety and relief valves. The pressure sensor concept proposed for ACNGS has been reviewed and approved by the NRC Staff for the Grand Gulf facility.

Q. Why is the system to be used at Allens Creek unambiguous?

A. When the safety/relief valve is closed, the pipe pressure should be near the containment pressure. However, as soon as the valve is opened, an almost instantaneous pressure rise would be detected by the sensor and would trigger the alarm. For the reasons noted in the previous answer, the thermocouple systems used in the past could give an indication that the valve was open when, in fact, it was closed.

Q. Does the pressure sensor to be used at Allens Creek provide a direct indication of valve position?

A. To respond to this question with precision, the pressure sensor provides a direct indication of <u>flow</u> through the valve. However, since flow is the parameter of interest, the design is acceptable to the Staff, and as noted above, a similar design has been approved by us previously for the Grand Gulf facility.

Q. How sensitive will the pressure sensors be with regard to detecting flow through safety/relief valves?

A. Typical pressure sensors are calibrated to read full flow, i.e. safety/relief valve fully open and reactor pressure in the normal operating range, and will not be very sensitive to low flows (flow less than approximately five percent of full flow). Small leakages may, therefore, not be indicated by the sensor. However, this small leakage is not significant from a safety standpoint. There is other instrumentation installed in the power plant which would allow the operators to deduce leakages above typical tech spec limits, i.e. 5gpm.

Q. Is there any concern over the feasibility of the position indication system to be used at Allens Creek?

A. No. The pressure increase in the pipe running from the safety/relief valve to the suppression pool will provide an easily measured and definite signal to the operators, alerting them to the status of the safety/relief valve. This type of pressure sensing device has had substantial successful operational use in both nuclear and non-nuclear facilities for many years.

Q. What is your conclusion on regarding this contention?

A. The system to be used at Allens Creek represents a significant improvement in the valve position indication system previously used in operating reactors. It will provide a direct indication of flow through the valve and thus an unambiguous indication of valve position to the operators. Further, there is no question that the system is feasible as demonstrated by the use of similar systems in different types of industrial facilities for many years.

Marvin W. (Wayne) Hodges

Professional Qualifications

Reactor Systems Branch

Division of Systems Integration

U. S. Nuclear Regulatory Commission

I am employed as a Section Leader in Section B of the Reactor Systems Branch, Division of Systems Integration.

I graduated from Auburn University with a BS degree in Mechanical Engineering in 1965. I received a Master of Science degree in Mechanical Engineering from Auburn University in 1967.

In my present work assignment at the NRC, I serve as principal reviewer in the area of boiling water reactor systems. I also participate in the review of analytical models use in the licensing evaluations of boiling water reactors and I have the technical review responsibility for many of the modifications and analyses being implemented on boiling water reactors after the Three Mile Island, Unit-2 accident.

As a member of the Bulletin and Orders Task Force which was formed after the TMI-2 accident, I was responsible for the review of the capability of BWR systems to cope with loss of feedwater transients and small break loss-of-coolant accidents.

I have also served at the NRC as a reviewer in the Analysis Branch of the NRC in the area of thermal-hydraulic performance of the reactor core. I served as a consultant to the RES representative to the program management group for the BWR Blowdown/Emergency Core Cooling Program.

Prior to joining the NRC staff in March, 1974, I was employed by E. I. DuPont at the Savannah River Laboratory as a research engineer. At SRL, I conducted hydraulic and heat transfer testing to support operation of the reactors at the Savannah River Plant. I also performed safety limit calculations and participated in the development of analytical models for use in transient analyses at Savannah River. My tenure at SRL was from June 1967 to March 1974.

From September 1965 to June 1967, while in graduate school, I taught courses in thermodynamics, statics, mechanical engineering measurements, computer programming and assisted in a course in the history of engineering. During the summer of 1966, I worked at the Savannah River Laboratory performing hydraulic testing.