UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of

HOUSTON LIGHTING & POWER COMPANY)) (Allens Creek Nuclear Generating) Station, Unit No.1))))	Docket No. 50-466

AFFIDAVIT OF RICHARD C. STIRN

State of California
County of Santa Clara

I, Richard C. Stirn, Manager of Core and Fuel System Design within the Nuclear Power Systems Engineering Department of the General Electric Company, of lawful age, being first duly sworn, upon my oath certify that the statements contained in the attached pages and accompanying exhibits are true and correct to the best of my knowledge and belief.

Executed at San Jose, California
July 29, 1980

Richard C. Stirn

Subscribed and sworn to before me this 29 day of July, 1980.

OFFICIAL SEAL
KAREN S. VOGELHUBER
NOTARY PUBLIC - CALIFORNIA
SANTA CLARA COUNTY
My Commission Expires Dec. 5, 1980

Karen S. Vogelhuber
NOTARY PUBLIC IN AND FOR SAID
COUNTY AND STATE

My commission expires 12-5 of 1980

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(Allens Creek Nuclear Generating Station, Unit No. 1)	§	

Affidavit of Richard C. Stirn

My name is Richard C. Stirn. I am employed at General Electric Company as a Professional Nuclear Engineer. I have been so employed for fifteen years. A statement of my experience and qualifications is set out in Attachment 1.

This affidavit addresses the issues raised in Doherty's Contention No. 24 which states that the Applicant has not provided a basis for showing that the reactivity insertion from any dropped control rod will be sufficiently small to prevent the peak energy yield from exceeding 280 calories/gram of fuel.

I. Introduction

For gross control of reactivity in the Allens Creek reactor, cruciform control blades are inserted between the fuel assemblies. These control blades, or rods, enclose smaller rods filled from boron carbide, a neutron absorbing material. The reactor is controlled by driving these control blades (177 for ACNGS) into the reactor core to reduce reactivity (and thus power) and withdrawing the rods to increase reactivity (and power). The control blades are moved by hydraulic drives which insert or retract the blades in small increments and continuously drive the blades in on a shutdown signal. The hydraulic drives are attached to the bottom of the pressure vessel. Each drive is attached to the bottom of a control blade by a special coupling.

II. The Control Rod Drop Event

There are many ways of inserting reactivity into a boiling water reactor. However, most of them result in a relatively slow rate of reactivity insertion and therefore pose no threat to reactor control. It is possible, however, that a rapid removal of a high worth1/ control rod could result in a potentially significant power excursion. The design basis accident dealing with rapid removal of a control rod is the rod drop accident.

The worst-case credible control rod drop accident for the ACNGS design is described as follows:

(a) Reactor is operating at 50 percent control rod density (half of the rods withdrawn in a checkerboard pattern). This pattern results in the highest incremental

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1/ Control rod worth is the measure of reactivity which will be added to the reactor if the rod is moved.

rod worth for the inserted rod since the four adjacent control blades are withdrawn.

(b) A fully-inserted control rod drive must sustain a complete break or disconnection from its cruciform control blade at or near the coupling.

(c) The blade must stick in the fully-inserted position as the rod drive is withdrawn.

(d) The blade falls by gravity to the position occupied by the rod drive after it is withdrawn.

Without regard to how the control rod blade drops, the worst case result is a blade falling unimpeded by its drive under the influence of gravity. The mechanism which causes a rod to drop is not a concern provided that the maximum rate of fall (reactivity insertion) is used in the analysis. The sequence of events and the approximate times of occurrence are as follows:

Event	Approximate Elapsed Time
(a) Control rod which provides maximum incremental worth becomes uncoupled.	_____
(b) Operator selects and withdraws the control rod drive of the uncoupled rod such that proper core geometry for maxijmum control rod worth exists.	_____
(c) Uncoupled control rod sticks in the fully inserted position.	_____

Event	Approximate Elapsed Time
(d) Control rod becomes unstuck and drops at a nominal measured velocity*/ plus, for conservatism, three standard deviations.	0
(e) Reactor goes on a positive period and initial power burst is terminated by the Doppler reactivity feedback.	1 sec.
(f) APRM 120 percent power signal scrams reactor.	____
(g) Scrams terminates accident.	5 sec.

This rod drop event sequence is the worst case because no other credible sequence of events can add positive reactivity at a faster rate.

III. Rod Pattern Control System

The purpose of the Rod Pattern Control System (RPCS) is to limit the worth of any control rod such that no unacceptable effects will result from a rod drop accident.

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*/ The velocity is limited by the control rod velocity limiter. The velocity limiter is in the form of two nearly mated conical elements that act as a large piston inside the control rod guide tube. The velocity limiter is provided with a streamlined profile in the scram (upward) direction. Thus, when the control rod is scrammed, water flows over the smooth surface of the upper conical element into the annulus between the guide tube and the limiter. In the dropout direction, however, water is trapped by the lower conical element and discharged through the annulus between the two conical sections. Because this water is jetted in a partially reversed direction into water flowing upward in the annulus, a severe turbulence is created, thereby slowing the descent of the control rod assembly to less than 5 feet/second at 70°F.

The RPCS will apply rod blocks before any rod motion can produce high worth rod patterns. The RPCS is a dual channel, safety-related system. These electronic circuits will have, in permanent storage, the identification of all rod groups and logic control information required to prevent movement of rods into unacceptable rod patterns. The RPCS, hence, limits the maximum rod worth of any rod which might uncouple and drop as discussed above.2/

IV. Consequences of Rod Drop Event

With the RPCS operational, the maximum incremental worth of any control rod is limited to approximately .8 percent ΔK. This limit is derived from the analysis in "Banked Position Withdrawal Sequence," NEDO-21231 (January, 1977). This very low incremental rod worth will produce a specific enthalpy well below the NRC design limit of 280 calories/gram. The peak enthalpy from a dropped rod given the conditions described above are less than 135 calories/gram. This result is computed using an adiabatic approximation of a super-prompt, critical large core and a two-dimensional, multi-group flux representation, as discussed in "Rod Drop Accident Analysis for Large Boiling Water Reactor," NEDO-10527 (March, 1972).

2/ The design and functioning of the RPCS system is more fully described in the Affidavit of Mr. J. F. Lesyna filed concurrently in this docket.

V. Conclusions

The design basis control rod drop event is the worst case reactivity insertion accident. To minimize its effect, a highly reliable Rod Pattern Control System has been designed for Allens Creek which will maintain individual incremental control rod worth less than .8 percent. As a result, a rod drop event cannot produce a specific fuel enthalpy greater than 135 calories per gram.

ATTACHMENT 1

QUALIFICATIONS

Richard C. Stirn, Manager
Core and Fuel Systems Design
General Electric, NEBG

My name is Richard C. Stirn. My business address is 175 Curtner Avenue, Mail Code 740, San Jose, California, 95125. I am a registered Professional Nuclear Engineer in the State of California (NU 630). As Manager I have the responsibility of directing core and fuel systems design for the General Electric Company, NEBG.

I graduated from Tennessee Technological University in 1962 where I received a Bachelor of Science Degree in Engineering Science. During the Summer of 1962 I worked for the Arnold Engineering Development Center in Tullahoma, Tennessee as an Engineer.

In the Fall of 1962 I entered Purdue University on an AEC Fellowship, and in August of 1964 I received a Master of Science Degree in Nuclear Engineering. Upon completion of my studies at Purdue I entered the University of Arizona to work toward a PhD degree in Nuclear Engineering; however, I left school in February of 1965 to work for the General Electric Company, NEBG before completing the PhD requirements.

Upon joining General Electric I entered the Engineering Training Program and had assignments dealing with light water moderated thermal reactors, steam cooled fast reactors, and sodium cooled fast reactors. After completing my training assignment in October 1967, I was appointed to the position of Technical Leader of Core Dynamics and Reactivity. In June 1972 I was appointed to the position of Manager of the Nuclear Safety Analysis Component of the Core Nuclear Engineering Unit. In this position I co-authored or contributed to three papers and three reports on the topic of nuclear reactor excursion analysis. I also participated in the development of the control rod drop accident boundary value approach for the reload licensing submittal.

In September of 1974 I assumed my present responsibilities as Manager, Core and Fuel Systems Design. In this capacity I am responsible for the development of system requirements for the Core and Fuel Perfomance, Core Performance Transient, and Fuel Mechanical Systems. Additional responsibilities include core thermal hydraulics evaluations, the development and issuance of core physics design requirements for reactivity control systems, the performance of criticality analyses of the fuel storage and handling facilities, and the development of core physics design bases for plant transients.

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