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 DONALD L. PETERSON

State of California
County of Santa Clara

I, Donald L. Peterson, Manager Control Rod Drive Line, 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.

Donald L. Peterson

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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Affidavit of Donald L. Peterson

My name is Donald L. Peterson. I am employed at General Electric Company as a Professional Nuclear Engineer, as I have been for 25 years. A statement of my experience and qualifications is set out in Attachment 1.

This affidavit responds to Intervenor Doherty's Contention No. 28 which postulates that a control rod can be ejected by containment pressure or by the pressure in the SCRAM Discharge Volume Tank (SDVT). Intervenor contends that such a rod ejection would create a more rapid reactivity insertion than a rod drop, the design basis reactivity insertion accident. As support for this assertion, Intervenor relies on the power excursion accident at the Stationary Low Power Reactor (SL-1) in January 1961.

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 with 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.

II. The Rod Drop Event

As testified to by Mr. Stirn in an affidavit also filed in this proceeding, it is possible that a rapid removal of a high worth[1/] 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:

1/ Control rod worth is the measure of reactivity which will be added to the nuclear reaction is the rod is removed.

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

b) The blade must stick in the fully inserted position as the rod drive is withdrawn.

c) The blade falls by gravity after the rod drive is fully withdrawn.

As explained by Mr. Stirn, the above described sequence assumes that the Rod Pattern Control System (RPCS) is operating. This is a reasonable assumption because the Rod Pattern Control System (RPCS) is a safety related, hard-wired, dual channel system which must operate to move control rods and which cannot be bypassed by the operator.[2/] The RPCS limits the amount of reactivity which may be inserted into the core by a dropped rod under the assumptions listed above. This is accomplished by physically preventing rod movement if the reactivity worth of the control rod chosen for movement by the operator is not in a proper preplanned sequence which minimizes rod worth.[3/]

2/ For a fuller description of the RPCS, see the affidavit of Mr. Lesyna, filed in this docket concurrently.

3/ The sequence is provided in detail on pages 7.7-8.8 of the ACNGS PSAR.

The enforced sequence assumes that the magnitude of the maximum control rod worth will be limited such that unacceptable fuel damage cannot occur in the event of a rod drop. Hence, with the RPCS operating, the peak fuel enthalpies that can result from the postulated drop of these rods has been shown to be well below the energy deposition (i.e., fuel peak enthalpy) design limit thereby resulting in inconsequential fuel damage.

III. The Hypothesized Rod Ejection Accident

According to Intervenor's contention, a control rod might be removed faster than has been assumed under the assumptions used for the rod drop event. This could theoretically produce higher fuel enthalpies (and thus worse fuel damage) than predicted. Intervenor hypothesizes that containment pressure and/or pressure from the SCRAM Discharge Volume Tank (SDVT) would act to drive the rod out faster than would gravity acting on a decoupled rod.

A. The Rod Control System

In order to understand Intervenor's conention, a simplified understanding of the Control Rod Drive (CRD) Hydraulic Control Units (HCU) is helpful. A simple illustration is attached as Exhibit A. The CRD is primarily a piston connected to the control rod and driven up or down by varying the pressure differential across the piston.

During a SCRAM, water at several hundred pounds per square inch above reactor vessel pressure automatically enters the CRD below the drive piston and pushes the rod into the core. In order to displace the water above the piston for such a rapid insertion, a special flow volume called the scram discharge volume is available. The scram discharge volume consists of header piping which connects to each control rod hydraulic control unit and drains into an instrument volume. The header piping is sized to receive and contain all the water discharged by the drives during a scram, independent of the instrument volume.

During normal plant operation the scram discharge volume is empty, and vented to atmosphere through an open vent and drain valve. When a scram occurs a signal from the Reactor Protection System closes these vent and drain valves to conserve reactor water. Lights in the control room indicate the position of these valves.

During a scram, the scram discharge volume partially fills with water displaced from above the drive pistons. Following the scram, the control rod drive seal leakage[4/] from the reactor continues to flow into the scram discharge

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4/ There are two leakage paths across the drive seals: one from the CRD supply pumps and another from reactor pressure.

volume until the discharge volume pressure equals the reactor vessel pressure. A check valve in each HCU prevents reverse flow from the scram discharge header volume to the drive. When the initial scram signal is cleared from the Reactor Protection System, the scram discharge volume signal is overridden with a key lock override switch, and the scram discharge volume is drained and returned to atmospheric pressure.

B. The Hypothetical Accident

As explained above, the SDV is at atmospheric pressure at all times except after a SCRAM. Hence, at these times the pressure from the SDV could not approach the value necessary to move any of the 177 CRD pistons. After a SCRAM, as the SDV is filled by seal leakage from the reactor, the pressure above the CRD piston is at all times equal to reactor pressure. Consequently, the fact that the SDV has pressurized to reactor pressure is of no consequence. It would add no additional pressure to drive the rod out of the core. Furthermore, since check valves are in all 177 lines leading to the SDV, any pressure held by the SDV could not be transmitted back to the CRD unit. Obviously, then, the pressure or lack of it in the SDV has no effect on the Control Rod Drives.

Intervenor has also suggested that containment pressure would add to or produce a rod ejection event. Since the Control Rod Drive units as well as their associated Hydraulic Control Units are sealed, there is no way in which the drive water would see containment pressure unless the system were ruptured. Since pressures in the CRD units are close to reactor operating pressure (several hundred psi) and containment design pressure is at a maximum 15 psi, the effect of a rupture on the control rod drive would be to relieve pressure above or below the drive piston. The release of pressure below the piston is discussed in detail below; the consequences are well accounted for. The release of pressure above the piston would drive the control rod into the core. Hence, containment pressure could not contribute to a rod ejection.

Although not covered by Intervenor's contention, one can postulate an event in which the pressure below the CRD drive piston is relieved by some incredible failure while reactor pressure acts above the pistons. However, even if the pressure below the drive piston were relieved with the drive latched (as it would normally be), no control rod withdrawal would occur. The CRD collet latch mechanism would hold the the CRD in place because the collet is held in the latching position with a heavy return spring and requires at least 100 psi above reactor pressure in the area over the piston to unlatch. If the pressure were relieved below the drive piston while the CRD was being moved (unlatched position), the hydraulic pressure which unlatches the CRD would drop and spring force would cause the collet to latch and stop rod withdrawal. Furthermore, even if the collet were assumed to jam in the open position, the steady-state control rod withdrawal velocity would be 2 ft/sec compared with 5 ft/sec for a rod drop event. This number was calculated by an analysis of the forces and hydraulic losses in the drive during this postulated event. Since lower withdrawal velocities result in lesser reactivity insertions (and thus lower peak enthalpies and potential fuel damage), this postulated event does not produce a rod ejection accident that compares with the rod drop event.

One other event which might be postulated is the failure of the drive housing such that the entire Control Rod Drive and housing becomes detached from the vessel. This scenario would result in the following sequence of events: The housing would separate from the vessel. The control rod drive and housing would be blown downward against the support structure by reactor pressure acting on the cross-sectional area of the housing and the drive. The downward motion of the drive and associated parts would be determined by the gap between the bottom of the drive and the support structure and by the deflection of the support structure under impact. The support structure is composed of a series of deep I beams located under the reactor vessel and attached to the reactor pedestal (not the reactor vessel). Suspended from these beams are tie rods attached to support blocks placed under the individual drives. Shock-absorbing springs at the tie rod/beam connection limit the maximum housing movement to approximately three inches under the loads anticipated from reactor pressure and driveline weight. In the current design, maximum deflection of the support structure after impact by the CRD housing is approximately 3 inches. If the collet were to remain latched, no further control rod ejection would occur and the housing would not drop far enough to clear the vessel penetration.

If the basic housing failure were to occur while the control rod is being withdrawn (this is a small fraction of the total drive operation time) and if the collet were to stay unlatched, the following sequence of events is possible: The housing would separate from the vessel. The drive and housing would be blown downward against the control rod drive housing support. Since in this instance the control rod is unlatched, the control rod will continue to withdraw after the control rod drive housing has been stopped by the drive housing support. The equivalent steady-state rod withdrawal velocity for this occurrence would be 0.3 ft/sec[5/] as compared with 5 ft/sec for the rod drop event. Clearly, this event also does not produce a rod ejection accident with consequences more adverse than the rod drop event.

The withdrawal speed for the instance where pressure below the CRD piston has been removed and the collet fails (2 ft/sec) is higher than the velocity for the case where the CRD housing has detached and the collet fails (.3 ft/sec) because in the former case reactor pressure above the CRD drives the rod out with no pressure below the drive piston. In the latter case pressure below the piston is not removed during the event. There are no other credible rod ejection scenarios with more adverse consequences because the failure of all known mechanisms which could produce uncontrolled withdrawl of control rods have been reviewed and analyzed.

III. <u>The SL-1 Event</u>

Intervenor has used the SL-1 accident to support the contention that a rod ejection event would produce a

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5/ This withdrawal velocity was calculated by the same means used to determine the withdrawal speed for the instance where pressure is relieved below the drive piston.

power excursion larger than a rod drop event. The SL-1 was a military pressurized water reactor which used control rods inserted into the top of the reactor. At the time of the accident, the reactor was shutdown (all controls rods in) and the control rod drives were undergoing maintenance. Results of investigation showed that a control rod, detached from its drive, had been removed to a point, where the reactor began a power excursion. As a result of the power excursion, control rods were blown out of the core and the reactor was demolished.

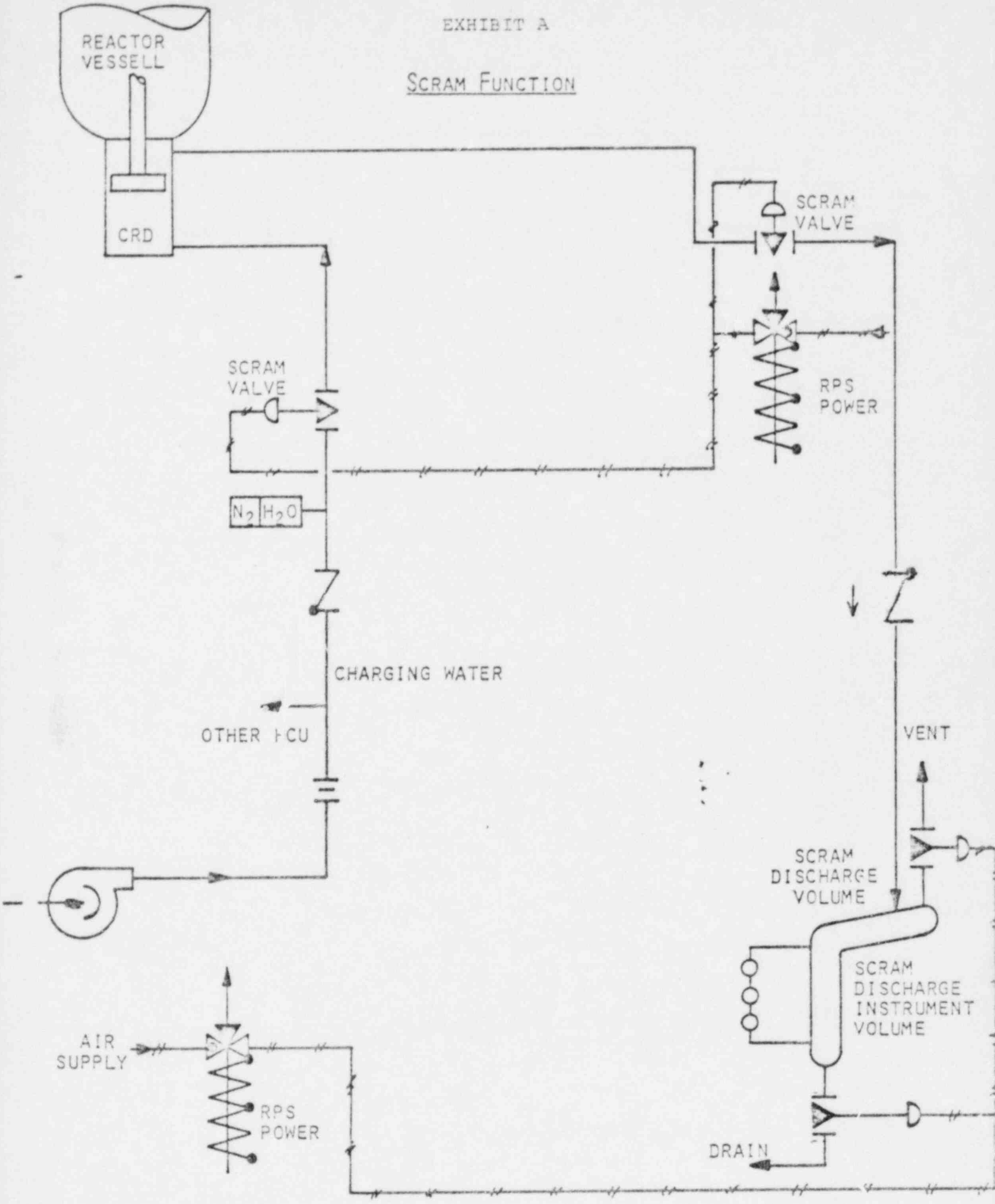
The SL-1 reactor and accident are in no way relevant to ACNGS. First, when the ACNGS reactor is shut down (all control rods in), the complete removal of one control rod would not bring the reactor critical. Secondly, rod ejection for ACNGS is physically prevented or the effects are bounded by the rod drop event. The SL-1 reactor was not so designed. Hence, Intervenor's reference to the SL-1 event to support this contention is totally inappropriate.

IV. Conclusion

After thoroughly analyzing the range of events which could rapidly remove a control rod from the reactor, it is apparent that the consequences of a design basis rod drop event are bounding.

EXHIBIT A

SCRAM FUNCTION



ATTACHMENT 1

PROFESSIONAL QUALIFICATIONS
DONALD L. PETERSON
CONTROL ROD DRIVE LINE DESIGN
GENERAL ELECTRIC COMPANY

My name is Donald L. Peterson. My business address is 175 Curtner Avenue, San Jose, California 95125. I am Manager, Control Rod Drive Line Design Unit for the Boiling Water Reactor System Department of the General Electric Company. In this position I am responsible for the design of components such as the control rod drive, control rod, hydraulic control unit and other components associated with the control rod drive system.

I have a Bachelors Degree in Mechanical Engineering from the University of Washington, Seattle.

After two years of service as a Gunnery Officer in the U.S. Navy and a short term as an Instructor in Mechanical Engineering Department of the University of Washington, I joined General Electric Company. Prior to transferring to General Electric Company's boiling water reactor operations, I worked at the Hanford operation for 7 1/2 years. My responsibilities included design of remote handling facilities, test facilities and control rod drive mechanisms.

For the past 25 years I have been responsible for the design of components for boiling water reactors, being the responsible design engineer for control rod drives on early General Electric reactors. I have continuously served as Technical Leader or Manager of the group responsible for control rod drives and associated components since that time.

I am a Professional Engineer in the State of California.