UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of

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

AFFIDAVIT OF PETER P. STANCAVAGE

State of California
County of Santa Clara

I, Peter P. Stancavage, Manager of Containment Engineering, within in the Domestic BWR Projects Department of 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.

Peter P. Stancavage

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

Ruthe M. Kinnamon
NOTARY PUBLIC IN AND FOR SAID
COUNTY AND STATE

My commission expires March 28 of 1981.

OFFICIAL SEAL
RUTHE M. KINNAMON
NOTARY PUBLIC - CALIFORNIA
SANTA CLARA COUNTY
My comm. expires MAR 28, 1981
175 Connor Ave., San Jose, CA 95125

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

Affidavit of Peter P. Stancavage

My name is Peter Stancavage. I am employed by General Electric Company as a nuclear and mechanical engineer. I have been employed in this capacity for 12 years. A statement of my experience and qualifications is set out in Attachment 1.

I. Introduction

The purpose of this affidavit is to address Mr. Doherty's Contention 5 which alleges that the control rod drive mechanism hydraulic control units (HCU) and the transversing in-core probe (TIP) may be damaged by the hydrodynamic forces of a high vertical water swell in the suppression pool following a loss-of-coolant accident (LOCA).[1/]

1/ LOCA is the sudden break of a high-energy pipe in the reactor coolant pressure boundary of the nuclear steam supply system. The largest possible break is the break of a main steam line.

II. Description of the Mark III Containment and Pool Swell Phenomena

The Allens Creek Nuclear Generating Station design uses a General Electric sixth generation boiling water reactor nuclear steam supply system with a third generation pressure suppression containment system. (This combination bears the name BWR/6--Mark III.) The basic Mark III containment design is shown in the attached diagram (Exhibit 1). The reactor primary system is surrounded by a cylindrical concrete drywell structure which is in turn surrounded by the primary containment. At the base of the drywell a series of horizontal open-ended pipes (vents) in three rows connects the drywell to the containment. The vents are submerged in an annular pool of water that is retained by a weir wall inside the drywell. Any steam released in the drywell from a postulated pipe break will be forced through the horizontal vents into the suppression pool where it will be condensed by the pool water.

Almost immediately following a postulated LOCA, the drywell is pressurized by reactor steam, and a mixture of steam and air is directed to the suppression pool through the horizontal vents. The rapid increase in drywell pressure will accelerate the water initially standing in the weir annulus and horizontal vents. Immediately following the clearing of standing water in any vent, drywell air and steam will form a bubble at the vent exit. This bubble will expand and depressurize to the local hydrostatic pressure. These bubbles cause an upper displacement of the pool water above the vents. The bubbles rise relative to the pool water, reducing the thickness of the water ligament or film above the bubbles. When the bubbles break through the water surface, a froth is formed which rises further before falling back into the suppression pool. The initial motion of the water film and the subsequent motion of the froth create impact and drag loads on equipment and platforms located above the pool surface. The entire process is referred to as "pool swell."[2/]

The pool swell loads on structures and components above the suppression pool have been evaluated in more than fifty full-scale and subscale experiments as part of the

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2/ Safety relief valve (SRV) actuation also introduces air into the pool as the released steam displaces the smaller air volume occupying the blowdown lines. However, SRV pool swell does not exist. Extensive in-plant tests, laboratory tests and an understanding of the phenomena involved in SRV discharge demonstrate that there is no pool swell due to this discharge. An understanding of the phenomena is acquired from scaling laws and analytical models of the SRV discharge. Full-scale in-plant tests were conducted at Monticello, Caroso, Tokai, KKB, KKP and Fukushima-6. Laboratory tests were also conducted by General Electric, KWV and CNEN. All these tests confirm that SRV pool swell does not occur.

Mark III test program conducted by the General Electric Company. From this information, loads are selected and used in the design of the ACNGS plant by the architect-engineer and in General Electric's analysis to qualify equipment supplied by General Electric.

III. Mark III Test Program

Immediately following the introduction of the BWR/6--Mark III, the General Electric Company started an extensive experimental and analytical effort to confirm the Mark III design. The purpose of the Mark III Confirmatory Test Program was to confirm the analytical methods used to predict the drywell and containment responses following a LOCA and to obtain information on the hydrodynamic loads that are generated in the vicinity of the suppression pool during a LOCA.

The General Electric Mark III containment pressure suppression testing program was initiated in 1971 with a series of small-scale tests. The test apparatus consisted of small-scale simulations of the reactor pressure vessel, drywell, suppression pool and horizontal vents. A total of sixty-seven blowdown runs were made. The purpose of these tests was to determine the behavior of the horizontal vents and to obtain data for determining the acceleration of the water in the test section vents during initial clearing. This information was used to establish an analytical model for predicting vent system performance in Mark III and the resulting drywell pressure response.

In November 1973, testing in the Mark III Pressure Suppression Test Facility (PSTF) began. The PSTF consists of an electrically heated steam generator connected to a simulated drywell which can be heated to prevent steam condensation within its volume during the simulated blowdowns. The drywell is modeled as a cylindrical vessel having a 10-foot diameter and 26-foot height. A 6-foot diameter vent duct passes from the drywell into the suppression pool and connects to the simulated vent system. Pool baffles are used to simulate a scaled or full-scale sector of a Mark III suppression pool.

The full-scale PSTF testing performed between November 1973 and February 1974 obtained data for the confirmation of the analytical model. In March 1974 pool swell tests were performed in the PSTF. These full-scale tests involved air blowdown into the drywell and suppression pool to identify bounding pool swell impact loads and breakthrough elevation, i.e., that elevation at which the water slug begins to break up and impact loads are significantly reduced. Impact load data were obtained on selected targets located above the pool. In June of 1974, after the PSTF vent and pool system was converted to 1/3-scale, four series of tests were performed to provide transient data on the interaction of pool swell with flow restrictions above the suppression pool surface.

The next series of 1/3-scale testing, which began in January, 1975, measured local impact pressures and total loads for typical small structures located over the pressure suppression pool including I-beams, pipes, and grating. Data from this test series expanded the data base from the full-scale air tests. A further series of 1/3-scale tests was added in June, 1975, to obtain comparable data on pool swell velocity and breakthrough elevation to the full-scale air tests.

The emphasis in the testing described above was directed at the evaluation of the pool swell phenomena. Each test run consisted of a simulation of the postulated blowdown transient. Various postulated break sizes up to two times the Design Basis Accident for the containment were tested. Data were recorded at selected locations around the test facility suppression pool throughout the blowdown so that the hydrodynamic conditions associated with each phase of the blowdown are known and are available for selecting appropriate design loading conditions. General Electric has used this data to develop hydrodynamic loading conditions in the GE Mark III reference plant pressure suppression containment system during the postulated LOCA.

IV. Pool Swell Loadings

Equipment and platforms, like the HCU, the HCU floors and the TIP, located in the containment annulus region above the pool surface experience pool swell induced dynamic loads, the magnitude of which are dependent upon both the location and the geometry of the surface exposed. The pool swell phenomenon occurs in two phases: "bulk" pool swell followed by a "froth" pool swell. Bulk pool swell imparts two different loads on exposed structures and components: impact loads and drag loads. The froth stage of pool swell contributes only a drag load.

A. Impact Loads

The PSTF air test data show that after the pool has risen approximately 1.6 times vent submergence below normal pool level (12 feet), the slug thickness has decreased to 2 feet or less and the impact loads are significantly reduced. For evaluating the time at which impact occurs at various elevations in the containment annulus, the maximum water surface velocity of 40 feet/second is assumed because this value bounds all the test data and analysis. The basis for the loading specification is the PSTF air test impact data. These tests involved charging the reactor simulator with 1000 psia air and blowing down through an orifice. Instrumented targets located over the pool provided the impact data.

For structures above the 18-foot elevation, the conservative froth impingement load is 15 psig based on data generated during the PSTF air test series. Again, this impingement load is applied uniformly to all structures.

B. Drag Loads

In addition to the impact loads, structures that experience bulk pool swell are also subject to drag loads as the pool water flows past them. Drag loads are calculated assuming a velocity of 40 feet/second between the pool surface and HCU floors.

C. Design of HCUs for Pool Swell Loads

Large platforms or floors will completely stop the rising pool, and thus incur larger loadings. For this reason, the HCU platform is located above the bulk pool swell zone. The GE Confirmatory Test Program indicates that pure bulk pool swell terminates at levels much lower than 18 feet above the suppression pool. Consequently, General Electric advises the architect-engineer to use 18 feet as the elevation of bulk pool swell with a linear transition from water to froth in the space of 18 feet to 19 feet above the normal pool surface. Therefore, for design application, the impact of water from bulk pool swell is applied conservatively at or below elevations of 19 feet above the surface of the suppression pool. The structures above this elevation experience an impulsive loading followed by a pressure differential loading. The impulsive load is due to the momentum of the froth which is decelerated by the structure. The pressure differential is based on an analysis of the transient pressure in the space between the pool surface and the HCU floor resulting from the froth flow through the approximately 1500 square feet vent area at this elevation. General Electric test results are the basis for the froth impingement load of approximately 15 psi lasting for 100 msec. An 11 psi froth flow pressure differential lasting for three seconds is based on an analysis of transient pressure in the space between the pool surface and the HCU floor. The approximate value of 11 psi is from a calculation which assumes that the density of the flow through the annulus restriction is a homogenous mixture of the top 9 feet of the suppression pool (i.e., 18.8 lb_m/ft^3). This is a conservative density assumption confirmed by the GE one-third scale test which shows an average density of approximately 10 lb_m/ft^3. The analytical model used to simulate the HCU floor flow pressure differential has also been compared with test data. These tests indicate HCU floor pressure differential is more realistically in the 3 to 5 psig range.

Vibratory response of the HCU floor to the froth impingement would subsequently transmit a load to the HCU

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modules. The magnitude of this load for Allens Creek will be computed by the architect-engineer in a plant unique dynamic analysis to assure that it does not exceed the dynamic qualification of the HCUs by General Electric.

D. Design of the TIP for Pool Swell Loads

General Electric PSTF tests demonstrate that for structures such as the TIP station, which is located approximately six feet above the suppression pool surface, pool swell impact loads are not experienced. The TIP station does experience a drag load and a "bubble" load. Bubble pressure load occurs when the air in the drywell is driven through the vents and forms air bubbles in the suppression pool prior to bulk pool swell. The pressure of these bubbles is then exerted on the wetted surfaces around the suppression pool.

PSTF data also establish that the TIP station would experience a maximum drag load of 11 psid and a 21.8 psid bubble pressure load. The TIP system itself is protected from the loads by cantilever structures which extend beneath the surface of the suppression pool and are specifically designed by the architect-engineer to absorb this loading.

In a larger sense, the issue of pool swell loading on the TIP station is a red herring. The TIP is a movable radiation source used to calibrate the Local Power Range Monitors when the reactor is shut down. It is not designed or

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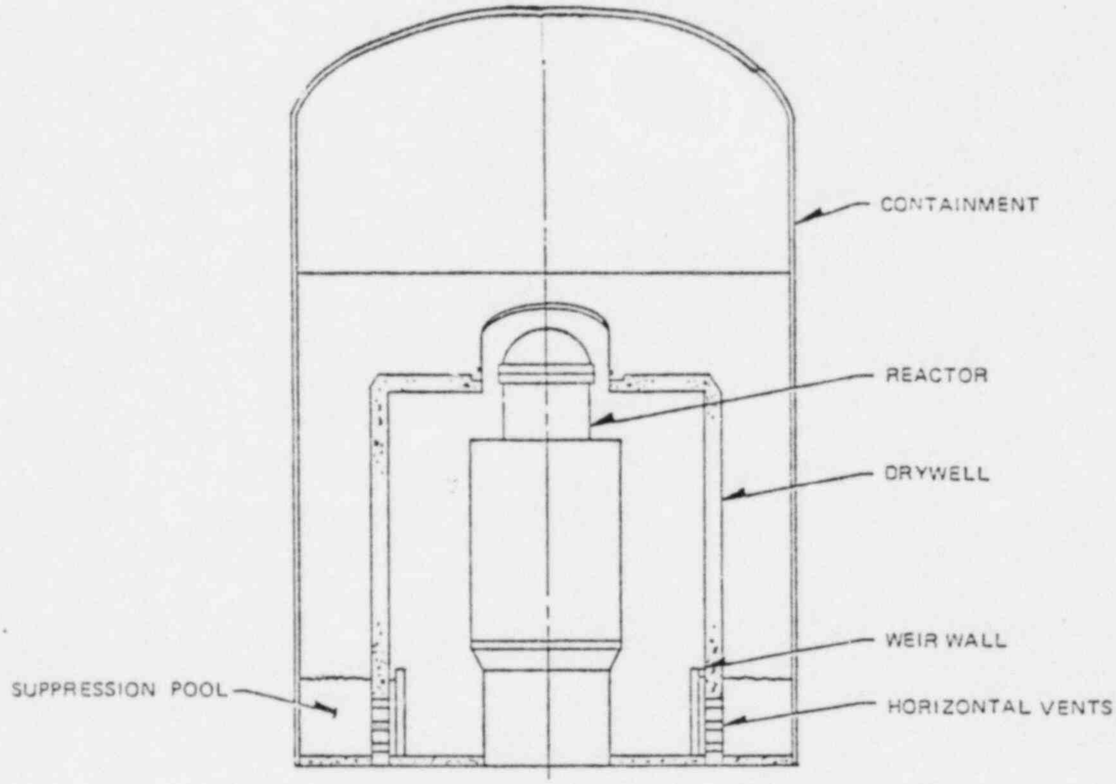
used to perform any safety function whatsoever. Consequently, its ability to survive a LOCA environment, including pool swell loading, has no importance save an economic effect which pales in comparison to the other consequences of such an accident.

EXHIBIT 1



Mark III Reactor Building

ATTACHMENT 1

PROFESSIONAL QUALIFICATIONS
PETER P. STANAVAGE
MANAGER - CONTAINMENT ENGINEERING

Mr. Stancavage has more than 13 years of Engineering experience with General Electric in the Nuclear Energy Group.

Mr. Stancavage is now the Manager of Containment Engineering, a position he has held for more than two years. His first eleven years with GE included a variety of Engineering jobs among which were three years in Containment Engineering, Radiological Evaluations and Nuclear Engineering.

Mr. Stancavage received his Master's Degree from M.I.T. in Nuclear Engineering. He completed his undergraduate work at U.S. Military Academy (West Point).