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 1)	§	
	§	

Statement of Material Facts As To
Which There Is No Genuine Issue
To Be Heard for McCorkle Contention 17

(1) The Allens Creek containment design does not allow 20 percent of the containment leakage to bypass the filtration systems. (Affidavit p. 7)

(2) A complete list of all potential leakage paths through containment penetrations was compiled (Exhibit A). From this list, six penetrations were identified which constitute potential unfiltered leakage paths (Exhibit B).

(3) Using the list of potential unfiltered leakage paths, the current best estimate of the maximum total unfiltered bypass leakage under LOCA accident conditions is .0195 percent per day of the containment volume. (Affidavit p. 4) The containment will be designed in any event to limit leakage to [illegible] percent by weight of the containment atmosphere per day at calculated peak pressure. (Affidavit p. 6).

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(4) Applicant will perform extensive pre-operational tests in accordance with 10 CFR Part 50, Appendix J, to assure that the containment will maintain its expected level of leak-tightness. (Affidavit p. 4-6).

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J M:gs

McCorkle Contention No. 17/
Filtration System Leakage

COST $ ______
PAID BY PLF. DEF.

UNITED STATES OF AMERICA NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY & LICENSING BOARD

IN THE MATTER OF:	)	
	)	
HOUSTON LIGHTING & POWER COMPANY	)	DOCKET NO. 50-466
(ALLENS CREEK NUCLEAR GENERATING	)	
STATION, UNIT 1)	)	

DEPOSITION OF:

BRENDA McCORKLE

International Court Reporters, Inc.

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191 / Bank of the Southwest Building • Houston, Texas 77002 • (713) 652-5911

that, then.

Okay. Turning to the last area that has to do with excessive leakage bypassing filtration systems, and I have again an introductory very broad question that I am forced to ask really. What leakage bypassing filtration system are you talking about? We can't pinpoint the structure or systems that you have reference to.

A Which interrogatory are you talking about?

Q That would be your interrogatory, the way I have it numbered, 17. I will read it the way I have it recorded. It says the containment as designed will allow excessive leakage to bypass the filtration system, power company admits that 20 percent of the leakage would not even be filtered and also the filter absorber, I think meant adsorber, may start a fire by auto ignition, if there is no water supplied by such auto ignition as required by the NRC regulation guide 1.52. That is the contention.

A I have that one down as my number 19.

Q Well, we'll straighten the numbering out later. But could you describe for me now what leakage from where to where bypassing what filtration systems are you talking about?

A I don't even remember. I have not looked at these since I wrote them or tried to answer your interrogatories.

Q Do you know whether or not you intended to reference normal operations or were you talking about the emergency conditions when you talked about leakage bypassing filtration systems?

A I don't remember that either.

Q Do you have --

A It sounds like it's talking about normal conditions. I think that I remember just discussing or rather Mr. Copeland asking questions on this and that they had, or HL&P had contained the 20 percent or they had gotten it to much less than 20 percent of the leakage on the filtering.

Q My next question was going to be where does the applicant admit that 20 percent of whatever leakage it is would not be filtered, that's another way of making entry into the documents so we can identify the structures or systems that you are talking about. Do you recall where--

A Page 629 of the safety report.

Q Of the original SER?

A I think so.

That has some reference, 20 percent of the total containment -- I think that Mr. Copeland said that this had been scaled down, in the last deposition.

Q So your concern in expressing this contention is based on the information recorded on page 629 of the SER?

A Right.

Q And those are the typical specification limit and the structures and systems you are concerned about, whatever is referenced on that page?

A Yes. I didn't come prepared to answer questions on that one.

Q Well, if you find that you need to amend or supplement something afterwards, give me a call and we'll try to work that out.

A Okay.

Q Afterwards. Let me ask you next, is it your concern that this unfiltered leakage which you have identified here will exceed part 100 or part 20 radiation dose limits? Now, it doesn't really say that in your contention, but something that I think is fairly easily inferred, it would help us if we could identify if that is the source of your concern?

A It could be, I don't remember anything about it.

Q Well, you see the thing that is missing is the yardstick against which to measure excessive. Once we have identified what leakage you are talking about, then we need an appropriate benchmark.

A That's true, but I have not done anything with this since I wrote these contentions, which was in November of '78, which was what, 18 months ago.

Q So, you don't know if it's part 100 limits you were talking about?

A No.

Q Would you agree that that would be an acceptable bench mark to measure this leakage?

A Yes.

Q If the applicant HL&P instituted redesign or whatever so that its releases were always below part 100 or part 20 limitations as is applicabl , would that remove your source of concern on this contention on leakage?

A If HL&P comes within the NRC guidelines, that's fine.

Q So long as NRC meets guidelines on this leakage, then you would be satisfied, is that correct?

A Yes.

Q I do need to clarify one other matter. It's sort of a separate part to this contention, related, but different than the leakage, the concerns you express over charcoal adsorbers. Am I to assume -- is it fair to assume that the charcoal adsorbers you are talking about would be in that system which is referenced on page 620 of the SER? In other words, there is a direct link between leakage past the filtration sistem and the filtration system that has the filter adsorber?

A I don't know.

Q Do you have any idea which filter adsorber or is it just any filter adsorber?

A I have no idea. It's been too long.

Q Well, the other clue is that you make reference to regulation guide 1.52, which governs

certain filtration systems and the charcoal adsorbers in those filtration systems.
Could it be that that is the clue that tells us which charcoal adsorber you are talking about?

A It could be, I just do not remember. It's been too long.

Q Well, do --

A I don't remember anything, ever getting interrogatories on that particular contention.

Q I'll find them for you.

A I've got them.

Q Well, we asked you in the second set of interrogatories, under Interrogatory C, Interrogatory 1.C, specific questions about unfiltered leakage and then you responded with answers regarding this leakage in part 100 doses.
This is your February 1st answers.

A Okay.

Q It has to do with this contention and our interrogatories on this.

A All right.

Q That's where I got the part 100 in the filter leakage. I think much earlier we asked some questions about the charcoal adsorber but not necessarily in that last go round.

A All right.

Q Let me ask you this. Until advised otherwise by yourself, can HL&P work on the presumption that the charcoal adsorber you were concerned about is the one referenced in regulation guide 1.52?

A Yes.

Q And in the same thing, if HL&P complies with regulation guide 1.52, would that remove the source of your concern?

A Yes.

Q One last question on this one. Are you familiar with Tex Pir~~t~~g's position on an identical or similar position on charcoal adsorbers?

A No, I haven't read any of them.

Q So there's no connection.

MR. BIDDLE: That is all I have. Thank you very much.

____

Brenda McCorkle

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 GUY MARTIN, JR.

State of New Jersey
County of Bergen

I, Guy Martin, Jr., Supervising Radiological Assessment Engineer, Allens Creek Project, for Ebasco Services Incorporated, of lawful age, being first duly sworn, upon my oath certify that I have reviewed and am thoroughly familiar with the statements contained in the attached affidavit addressing intervenor Brenda McCorkle's Contention 17 regarding filtration system leakage. All statements contained therein, which relate to Ebasco Services Incorporated scope of supply for the Allens Creek Nuclear Generating Station, are true and correct to the best of my knowledge and belief.

Subscribed and sworn to before me this ____ day of __________, 1980.

CAROL A. OPITENOK
NOTARY PUBLIC OF NEW JERSEY
MY COMMISSION EXPIRES SEPT. 18, 1983

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 WALTER F MALEC

State of New Jersey
County of Bergen

I, Walter F Malec, Supervising Mechanical Nuclear Engineer, Allens Creek Project, for Ebasco Services Incorporated, of lawful age, being first duly sworn, upon my oath certify that I have reviewed and am thoroughly familiar with the statements contained in the attached affidavit addressing intervenor Brenda McCorkle's Contention 17 regarding filtration system leakage and that all statements contained therein are true and correct to the best of my knowledge and belief.

Walter F Malec

Subscribed and sworn to before me this ______ day of ____________, 1980.

CAROL A. OPITENOK
NOTARY PUBLIC OF NEW JERSEY
MY COMMISSION EXPIRES SEPT. 13, 1983

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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 GUY MARTIN, JR.
AND WALTER F. MALEC

My name is Guy Martin, Jr. My business address is Two World Trade Center, New York, N. Y. I am the Supervising Radiological Assessment Engineer for the Allens Creek Project employed by Ebasco Services Incorporated. The statement of my background and qualifications is attached as Exhibit I to this testimony.

My name is Walter F. Malec. My business address is 160 Chubb Avenue, Lyndhurst, N. J. I am the Supervising Mechanical Nuclear Engineer for the Allens Creek Project employed by Ebasco Services Incorporated. The statement of my background and qualifications is attached as Exhibit II to this testimony.

This affidavit addresses the issues raised in McCorkle Contention No. 17. The contention states that the Allens Creek containment as designed will allow 20 percent of the containment leakage to bypass the filtration systems.

I. Introduction

The Allens Creek containment consists of a free-standing steel shell 1 1/2 to 1 3/4 inches thick which encloses the reactor vessel holding the reactor fuel. The containment is designed to protect the public from the release of radioactive fission products by providing a leak-tight barrier. However, for practical purposes, the containment must be penetrated by piping and other openings. Although these penetrations are sealed by some means such as redundant valving, a certain quantity of leakage is inevitable. NRC regulations (10 CFR, Part 50, Appendix J) limit the quantity of leakage allowed.

II. Containment Leakage Expected for Allens Creek

The Containment Vessel is a seismic Category I steel shell designed to confine the radioactive materials, gases under pressures and temperatures associated with a loss-of-coolant accident and all other abnormal operating conditions. The design leak rate will be 0.5 percent by weight of the contained atmosphere per day at calculated peak pressure. The Containment Vessel will be designed to contain any leakage from the drywell and the noncondensable gases from reactor vessel blowdown by the safety/relief valves or from the rupture of the largest pipe inside the drywell.

To determine the type of leakage which can be expected, a list of all potential leakage paths through containment penetrations was compiled (Table 6.2-12a of the Preliminary Safety Analysis

Report). This list is reproduced as Exhibit A. From this list, only six penetrations constitute potential unfiltered leakage paths. These six penetrations are listed in Table 6.2-13 of the PSAR and the table is reproduced as Exhibit B.

In arriving at the list contained in Exhibit B, an evaluation was made of all lines which penetrate the containment to determine the number and types of barriers to bypass leakage provided for each line. The types of bypass leakage barriers considered were as follows:

(a) Isolation valve outside containment.

(b) Isolation valve inside containment.

(c) Closed Category I piping system inside containment.

(d) Closed Category I piping system outside containment.

(e) Water seal in line.

(f) Line beyond isolation valve outside containment vented to annulus for filtration by the Standby Gas Treatment System (SGTS).

(g) Line terminates outside containment in filtered ECCS Area of Auxiliary Building.

Leakage barriers of types (c) through (g) effectively eliminate any bypass leakage. Leakage barriers of types (a) or (b) limit but do not eliminate bypass leakage. Therefore, lines containing any of the bypass leakage barriers (c) through (g) were not considered as potential bypass leakage paths. Lines containing only types (a) or (b) were included in Exhibit B as potential unfiltered leakage paths.

III. Unfiltered Leakage

The amount of containment leakage allowed in the Technical Specifications will be significantly less than that which would produce total off-site doses equal to the 10 CFR 100 limits. The contributors to this total leakage include the Standby Gas Treatment System releases, leakage to the controlled ventilation ECCS area of the Auxiliary Building and all unfiltered bypass leakage. The actual value of the bypass leakage technical specification will be determined as a result of LOCA dose calculations performed when the FSAR is prepared for submittal. However, a value of .0195 percent/day of the containment volume is the present best estimate of the maximum total unfiltered bypass leakage based on preliminary LOCA dose calculations. These dose calculations are provided in detail in Section 15 and Appendix 15.A of the PSAR.[1/]

IV. Tests and Inspections

In order to assure that the containment will maintain its expected level of leak-tightness, Applicant will conduct a leak testing program in accordance with

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1/ The fraction of total containment leak rate technical specification which will be released via potential bypass leakage lines is quoted at PSAR, p. 15.A-4b as 2.9×10^{-2}. This number is a typographical error. The correct value is 3.9×10^{-2}.

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Appendix J of 10 CFR 50. As required by Appendix J, three types of tests will be performed:

Type A - This test will measure the primary reactor containment overall integrated leakage rate. It will be conducted after the containment is completed and ready for operation and again about once every three and one-third years thereafter. In addition, any major modification or replacement of components of the primary reactor containment performed after the initial leak rate test shall be followed by either a Type A test or a Type B test of the area affected by the modification.

Type B - Appendix J defines these tests as those:

> intended to detect local leaks and to measure leakage across each pressure-containing or leakage-limiting boundary for the following primary reactor containment penetrations:
>
> 1. Containment penetrations whose design incorporates resilient seals, gaskets, or sealant compounds, piping penetrations fitted with expansion bellows, and electrical penetrations fitted with flexible metal seal assemblies.
>
> 2. Air lock door seals, including door operating mechanism penetrations which are part of the containment pressure boundary.
>
> 3. Doors with resilient seals or gaskets except for seal-welded doors.
>
> 4. Components other than those listed above which must meet the acceptance criteria in III.B.3 of Appendix J.

Except for containment air locks, Type B tests will be conducted during each reactor shutdown for major fuel reloading but in no case at intervals greater than two years. The seals of the personnel air locks will be tested after each opening or, if left unopened, at an interval not to exceed one year.

Type C - Type C tests are those intended to measure containment isolation valve leakage rates. The containment isolation valves included are those that:

1. Provide a direct connection between the inside and outside atmospheres of the primary reactor containment under normal operation, such as purge and ventilation, vacuum, relief, and instrument valves;

2. Are required to close automatically upon receipt of a containment isolation signal in response to controls intended to effect containment isolation;

3. Are required to operate intermittently under post-accident conditions; and

4. Are in main steam and feedwater piping and other systems which penetrate containment of direct-cycle boiling water power reactors.

Type C tests shall be performed for isolation valves during each reactor shutdown for major refueling.

V. Conclusion

The Allens Creek containment will be designed to limit leakage to 0.5 percent by weight of the containment atmosphere per day at calculated peak pressure. Applicant has calculated that, under loss of coolant accident conditions, a maximum of .0195 percent per day of containment volume may escape via the potential bypass leakage lines and that the resulting doses will not exceed the limits of 10 CFR Part 100. Hence, Intervenor's claim that 20 percent of the containment leakage will bypass filtration systems does not reflect the present plant design and the updated bypass leakage fraction calculations contained in PSAR, Section 15 and Appendix 15.A. Finally, the projected containment integrity will be assured by performing the leak-rate tests called for by 10 CFR, Appendix J.

EXHIBIT A

EVALUATION OF POTENTIAL
BYPASS LEAKAGE FOR CONTAINMENT
PENETRATIONS

System Service	Line Size (in.)	Bypass Leakage Barriers*	Considered Potential Bypass Path
Main Steam Lines A, B, C, and D	26	A, B, H	No
Feedwater A and B	20	A, B, E	No
RHR Pump A, B, and C Suction from Suppression Pool	24	A, D, E, G	No
RHR Shutdown Suction From Recirculation Loop	20	A, B, D, E, G	No
RHR Return A and B to Recirculation Loop	12	A, B, D, E, G	No
RHR A, B, and C LPCI	12	A, B, D, E, G	No
RHR A, B, and C Pump Test Lines to Suppression Pool	18	A, D, E, G	No
HPCS Pump Suction from Suppression Pool	24	A, D, E, G	No
HPCS Pump Discharge	12	A, B, D, E, G	No
HPCS Test Line to Suppression Pool	12	A, D, E, G	No
HPCS Minimum Flow Line	4	A, D, E, G	No
LPCS Pump Suction from Suppression Pool	24	A, D, E, G	No
LPCS Pump Discharge to Pressure Vessel	12	A, B, D, E, G	No
LPCS Test Line		A, D, E, G	No

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EXHIBIT A

System Service	Line Size (in.)	Bypass Leakage Barriers *	Considered Potential Bypass Path
Steam Supply the RCIC Turbine and RHR Heat Exchanger	10	A, B, D	No
RCIC and RHR to Head Spray	6	A, B, D, E	No
RCIC Pump Suction from Suppression Pool	6	A, D, E	No
RCIC Turbine Exhaust to Suppression Pool	12	A, D	No
RCIC Pump Discharge Minimum Flow Bypass	2	A, D, E	No
RCIC Vacuum Pump Discharge	2	A, G	No
CRD Pump Discharge	2	A, B, E	No
Station Air Supply	2	A, B	Yes
Instrument Air Supply	2	A, B	Yes
Reactor Building Closed Cooling Water Supply	14	A, B, E	No
Reactor Building Closed Cooling Water Return	14	A, B, E	No
Reactor Water Clean-up to Condenser and Radwaste	4	A, B, E	No
Reactor Water Clean-up Backwash Transfer Pump Discharge	4	A, B, E	No
Main Steam Drains to Condenser	3	A, B, E	No

EXHIBIT A

System Service	Line Size (in.)	Bypass Leakage Barriers *	Considered Potential Bypass Path
LPCS Minimum Flow Line	4	A, D, E, G	No
RHR Pump Minimum Flow Line (Typ 3)	4	A, D, E, G	No
Chilled Water System Supply	4	A, B, E	No
Chilled Water System Return	4	A, B, E	No
Containment Purge Supply	4	A, B, F	Yes
Hydrogen Purge Exhaust	4	A, B, D	No
Containment Vacuum Relief A and B	18	A, B, F	No
Fuel Transfer Tube	32	A, B, E	No
Demineralized Water Supply to Containment	4	A, B, E	No
Discharge from Fuel Pool Cooling and Cleanup to Containment Pool	6	A, B, E	No
Inlet to Fuel Pool Cooling and Cleanup from Containment Pool	10	A, B, E	No
Condensate Makeup Supply	2	A, B, E	No
Drywell Floor Drain Discharge Header	3	A, B, E	No
Containment Floor Drain Discharge	3	A, B, E	No

EXHIBIT A

System Service	Line Size (in.)	Bypass Leakage Barriers*	Considered Potential Bypass Path
Containment Ventilation Air Supply and Exhaust	36	A, B, F	No
Drywell Containment Equipment Drains	3	A, B, E	No

* Possible Bypass Leakage Barrier Designation :

A. Isolation valve outside containment
B. Isolation valve inside containment
C. Closed Category I piping system inside containment
D. Closed Category I piping system outside containment
E. Water seal in line
F. Line beyond isolation valve outside containment vented to annulus
G. Line terminates outside containment in filtered ECCS area of auxiliary building

EXHIBIT B

POTENTIAL UNFILTERED CONTAINMENT
BYPASS LEAKAGE PATHS

Description	Line Size (in)
Station Air Supply	2
Instrument Air Supply	2
Containment Purge Supply (2)	4
Main Steam Line Guard Pipe	
Feedwater Line Guard Pipe	
Personnel Air Lock	

EXHIBIT I

GUY MARTIN, JR
Supervising Engineer
Radiological Assessment

SUMMARY OF EXPERIENCE (Since 1965)

Total Experience - Fifteen years participation in Safety Analysis Reports, Environmental Reports, SAR amendments, licensing documents, and cost analysis for insurance premium determination.

Professional Affiliations - American Society of Mechanical Engineers
Health Physics Society
American Nuclear Society
Intern Engineer in New York State, Certificate No. 022127

Education - MS, Polytechnic Institute of New York, 1976 Nuclear Engineering
BE, City College of the City of New York,
School of Harvard University School of Public Health, 1977 - Radiological Surveillance Course.

REPRESENTATIVE EBASCO PROJECT EXPERIENCE (Since 1973)

Supervising Engineer

Participate in the coordination, technical review and preparation of Safety Analysis Reports (SAR), Environmental Reports (ER), SAR amendments and other licensing documents (e.g., Appendix I to 10 CFR 50 studies) for submittal to the Nuclear Regulatory Commission as part of the application for Construction Permit and Operating License of nuclear power plants.

Areas of complete responsibility include sections of the SAR dealing with the radiological dose assessment work associated with normal and hypothetical accident conditions. In this regard, conduct safety reviews of systems, specifications and operation from a nuclear safety viewpoint and check their compliance with established nuclear safety criteria.

Furnish technical support in the preparation of testimonies for safety hearings and ACRS presentation. Study, develop, maintain and use appropriate methods, including computer programs for evaluating radiological exposures.

PRIOR EXPERIENCE (8 years)

Equitable Life Assurance Society of the US
Cost Analyst

Work involved calculating and analyzing cost of various activities performed throughout the company; assisting departmental managers in their budget preparation work. Made statistical studies for determination of activity costs and providing company's actuaries support information for premium determination.

Dividend Specialist

Reviewed and analyzed dividend and claim reserve calculations. Prepared disbursement authorizations and dividend information reports for policy holders. Participated in training programs for new employees.

Publications

Martin, G and J Thomas 1978. Meeting the dose requirements of 10 CFR 100 for site suitability and general design criteria 19 for control room habitability: a parametric approach. Transactions of American Nuclear Society 24th Annual Meeting, Vol. 28.

Martin, G, D Michlewicz and J Thomas 1978. Fission 2120: a program for assessing the need for engineered safety feature grade air cleaning systems in post - accident environment. Proceedings of 15th DOE Nuclear Air Cleaning Conference.

Letizia, A P, G Martin and J F Silvey 1979 . - Implications for nuclear facilities of changes being initiated in the NRC standard atmospheric diffusion model. Proceeding of the 41st Annual Meeting of the American Power Conference.

Bhatia, R K, Mauro, J, Martin, G. Effects of Containment Purge on the Consequences of a Loss-of-Coolant Accident. Transactions of American Nuclear Society 1980 Annual Meeting.

Born — Philadelphia, Pennsylvania

Education — Polytechnic Institute of Technology, degree of Engineer in Nuclear Engineering - 1978
Massachusetts Institute of Technology, MS in Nuclear Engineering - 1970
U.S. Coast Guard Academy, BS - 1968

Member — American Nuclear Society

Licensed — Registered Professional Engineer in the State of New York (No. 56673)

Experience:

1980 — Ebasco Services Incorporated, Lyndhurst (NJ) Office; Supervising Engineer, Mechanical-Nuclear Engineering Department:

Houston Lighting & Power Co - Allens Creek NGS - Unit No. 1 - 1200 MW(e) BWR

Technical and administrative responsibility for mechanical, fire protection, plumbing, HVAC, stress analysis, hangers and supports, and inservice inspection activities. Includes schedules, budgets, and client relations.

1978-1980 — Ebasco Services Incorporated, Lyndhurst (NJ) Office; Principal Engineer, Mechanical-Nuclear Engineering Department

Houston Lighting & Power Co - Allens Creek NGS - Unit No. 1 - 1200 MW(e) BWR, Lead NSSS Engineer

Responsible for preparation and maintenance of ECCS and BOP flow diagrams, piping layouts, system design descriptions, inservice inspection provisions, Nuclear Island building general arrangements, PSAR and FSAR preparation, equipment sizing and specification, NSSS vendor interface for correspondence, drawing review, and contract administration.

1976-1978 — Ebasco Services Incorporated, New York Office; Senior Engineer, Mechanical-Nuclear Engineering Department including:

Houston Lighting & Power Co - Allens Creek NGS - Unit No. 1 - 1200 MW(e) BWR, Lead NSSS Engineer

Louisiana Power & Light Co - Waterford SES Unit No. 3 - 1165 MW(e) PWR. Lead NSSS Engineer

(Same responsibilities as listed for 1978-1980 above.)

1976-1978 (Cont'd)	Responsible for preparation and maintenance of ECCS and BOP flow diagrams, piping layouts, system design descriptions, inservice inspection provisions, Nuclear Island building general arrangements, PSAR and FSAR preparation, equipment sizing and specification, NSSS vendor interface for correspondence, drawing review, and contract administration.

* * * * *

1974-1976	United States Coast Guard, Marine Inspection Office, New York; Lieutenant - Supervisory Boiler Inspector. Responsibility for supervision, assignment and training of Marine Inspectors in largest Marine Inspection Office in country. Inspection of hull and machinery material condition of U.S. flag and foreign merchant vessels, and pressure vessels under construction. Application of various laws and regulations of the United States, ASME Code, ANSI, TEMA, NEC and NFPA Standards. Review of engineering plans and alterations, reports from field and resident inspectors.
1973-1974	United States Coast Guard, USCGC Spencer (WHEC-36), Lieutenant - Chief Engineer. Responsibility for operation, maintenance and repair of hull and engineering plant of 6200 slip twinscrew steamship. Direct supervision of 40 officers and men. Duties included preparation of repair specifications and maintenance of vessel records. Received Coast Guard Achievement Medal for superior performance of duty.
1970-1973	United States Coast Guard, Marine Inspection Office, New York, Lt and Ltjg - Marine Inspector. Inspection of hull and machinery of U.S. and foreign flag merchant vessels.
1968-1969	United States Coast Guard, USCGC Mellon (WHEC-717), Ensign, Assistant Engineer Officer.

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

CERTIFICATE OF SERVICE

I hereby certify that copies of Applicant's Motions for Summary Disposition dated August 4, 1979 in the above-captioned proceeding were served on the following by deposit in the United States mail, postage prepaid, or by hand-delivery this 4th day of August, 1980.

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Atomic Safety and Licensing
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U.S. Nuclear Regulatory
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C. Thomas Biddle Jr

C. Thomas Biddle, Jr.