Before the

UNITED STATES NUCLEAR REGULATORY COMMISSION

Docket No. 50-466

Allens Creek Nuclear Generating Station Unit 1

Amendment 53 to the

PSAR

Houston Lighting & Power Company, applicant in the above captioned proceeding, hereby files Amendment 53 to the Preliminary Safety Analysis Report filed in connection with its application.

Amendment 53 consists of additional PSAR information updating the PSAR since the issuance of the Allens Creek Safety Evaluation Report, Supplement 2.

Respectfully submitted

HOUSTON LIGHTING & POWER COMPANY

E. A. Turner

E. A. Turner
Vice President
Power Plant Construction
& Technical Services

369033

7909180368

STATE OF TEXAS

COUNTY OF HARRIS

E. A. TURNER, being first duly sworn, deposes and says: That he is Vice President of HOUSTON LIGHTING & POWER COMPANY, an Applicant herein; that the foregoing amendment to the application has been prepared under his supervision and direction; that he knows the contents thereof; and that to the best of his knowledge and belief said documents and the facts contained therein are true and correct.

DATED: This 17th day Sept, 1979.

Signed: E A Turner
E. A. Turner

Subscribed and sworn to before me
this 17 day of Sept, 1979.

Rita P. Villanueva
Notary Public in and for the
County of Harris, State of Texas

My commission expires:
4-30-81

960094

Houston
Lighting
& Power
Company

Electric Tower
P.O. Box 1700
Houston, Texas 77001

September 17, 1979
AC-HL-AE-357

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Sir:

Allens Creek Nuclear Generating Station
Unit 1
Docket No. 50-466
Amendment 53

Please find under separate cover sixty (60) copies of Amendment 53 to the Houston Lighting & Power Company Allens Creek Nuclear Generating Station Unit 1 PSAR. A copy of this transmittal letter is attached to each amendment copy.

Amendment 53 consists of additional PSAR information updating the PSAR since the issuance of the Allens Creek Safety Evaluation Report, Supplement 2.

Very truly yours,

E A Turner

E. A. Turner
Vice President
Power Plant Construc.ion
& Technical Servic·s

LDR/ngb

cc: J. G. Copeland (Baker & Botts)
R. Gordon Gooch (Baker & Botts)
J. R. Newman (Lowenstein, Newman, Reis, Axelrad & Toll)
P. A. Horn
All Parties

960095

Before the

UNITED STATES NUCLEAR REGULATORY COMMISSION

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960096

STATE OF TEXAS

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E. A. TURNER, being first duly sworn, deposes and says: That he is Vice President of HOUSTON LIGHTING & POWER COMPANY, an Applicant herein; that the foregoing amendment to the application has been prepared under his supervision and direction; that he knows the contents thereof; and that to the best of his knowledge and belief said documents and the facts contained therein are true and correct.

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My commission expires:
4-30-81

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ACNGS-PSAR

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TABLE 1.3-5

COMPARISON OF STRUCTURAL DESIGN REQUIREMENTS

	Allens Creek	Susquehanna	Bailly	Limeric
Seismic Design				
Reference PSAR Section	3.7	PSAR Appendix C	2.5.3	2.5.3
50% Safe Shutdown Earthquake (horizontal g)	0.05	0.05*	0.10*	0.06*
Safe Shutdown Earthquake** (horizontal g)	0.10		0.20	0.12
Earthquake vertical shock (% of horizontal)	67	60	67	66
Wind Design				
Reference PSAR Section	3.3	PSAR Appendix C Section C.2	12.2	2.3
Maximum sustained wind (mph)	157***	80	90	90
Tornadoes (mph)	290 rotational +70 trans. 360 tangential	300	300 tang. + 60 trans.	300

53 (C)

* previously called Operating Basis Earthquake

** previously called Design Basis Earthquake

*** 1 minute average at 30 foot level

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Spent fuel is stored under water in the Spent Fuel Pool. The racks in which spent fuel assemblies are placed are designed and arranged to ensure subcriticality in the storage pool. Spent fuel is maintained at a subcritical multiplication factor K_{eff} of less than or equal to 0.95 for all conditions. 53(D)

Refueling interlocks include circuitry which senses conditions of the refueling equipment and the control rods. These interlocks reinforce operational procedures that prohibit making the reactor critical. The Fuel Handling System is designed to provide a safe, effective means of transporting and handling fuel and is designed to minimize the possibility of mishandling or maloperation.

The use of geometrically safe configurations for new and spent fuel storage and the design of fuel handling systems precludes accidental criticality in accord with Criterion 62.

For further discussion, see the following sections:

a.	All other Instrumentation Systems Required for Safety	7.9
b.	Fuel Storage and Handling	9.1

3.1.2.6.4 Criterion 63 - Monitoring Fuel and Waste Storage

Appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas, (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels, and (2) to initiate appropriate safety actions.

3.1.2.6.4.1 Evaluation Against Criterion 63

Appropriate systems will be provided to meet the requirements of this criterion. A malfunction of the Fuel Pool Cooling and Cleanup System which could result in loss of residual heat removal capability and excessive radiation levels is alarmed in the Control Room. Alarmed conditions include high/low fuel pool cooling water pump discharge pressure and high/low level in the fuel storage pool and drain tanks. System temperature is also continuously monitored in the Control Room. Radiation Monitor continuously monitors radioactivity in this area and initiates an alarm in the Control Room on abnormal radiation levels. 35(C)

Radiation and tank and sump levels are monitored and alarmed to give indication of conditions which may result in excessive radiation levels in radioactive waste system areas. These systems satisfy the requirements of Criterion 63.

(C) Consistency
(D) Design

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TABLE 3.2-1 (Cont'd)

Principal Component[a]	Scope[b] of Supply	Safety[c] Class	NRC[d] Quality Group	Component[e] Location	ENVIRONMENTAL CAPABILITY: Seismic[f] Category	Extreme[g] Wind	Tornado/[h] Missile	Flood[i] Protection	Quality[j] Assurance Program	Comments	
3. Fuel Handling Building, Including:											
Base Slab	P	3	NA	M	I	b	b	a	B		
Structural Walls	P	3	NA	R	I	a	a	b	B		22
Structural Floors	P	3	NA	R	I	b	b	c	B		
Spent-fuel pool	P	2	NA	R	I	b	b	c	B		
Fuel cask storage pool	P	3	NA	R	I	b	b	c	B		
Spent fuel-storage racks	P	3	NA	R	I	b	b	c	B		
Temporary fuel-storage racks	GE	3	NA	C	I	b	b	c	B	53 (D)	
Fuel speciality racks	GE	3	NA	R	I	b	b	c	B		
Fuel Handling Crane	GE	3	NA	R	I	b	b	c	B		
All Seismic Category I equipment supports	P	3	NA	R	(u)	b	b	c	B		22
4. Control Building											
Base Slab	P	2	NA	M	I	b	b	a	B		22
Structural Walls	P	2	NA	B	I	a	a	b	B		
Structural Doors	P	2	NA	B	I	b	b	c	B		
All Seismic Category I equipment supports	P	1,2,3	NA	B	I	b	b	c	B	(v)	
5. Diesel Generator Building											
Base Slab	P	3	NA	M	I	b	b	a	B	13	22
Structural Walls	P	3	NA	S	I	a	a	b	B		
Structural Floors	P	3	NA	S	I	b	b	c	B		
All Seismic Category I equipment supports	P	3	NA	S	I	b	b	c	B		

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6.2.5.1 Design Bases

The following are the criteria used to design the Combustible Gas Control System:

a) The system will be designed in accordance with Regulatory Guide 1.7 Revision 1, September 1976 and General Design Criterion 41 of 10 CFR 50 Appendix A. 26

b) The hydrogen resulting from metal-water reaction is assumed to be the larger of the amounts which would evolve based on two criteria

i) A core wide reaction of the cladding to a depth of 0.00023 inch 53(U)

ii) Five times the percentage of total cladding mass reacted based on ECCS evaluation

In either case, the hydrogen is assumed to evolve in the first two minutes following the postulated LOCA.

c) The system will have the capability of sampling and measuring the hydrogen concentration throughout the drywell and containment during all modes of operation.

d) The system will have the capability of mixing the atmosphere in the containment and drywell following a LOCA.

e) The system will have the capability of controlling combustible gas concentrations in the containment atmosphere without reliance on purging and without the release of radioactive material to the environment.

f) The Combustible Gas Control System and the equipment for mixing, measuring and sampling will meet the design, quality assurance, redundancy, energy source, and instrumentation requirements compatible with the safety of the system.

g) The system will not introduce safety problems that would affect containment integrity.

h) As a backup to the hydrogen recombiner subsystem of the Combustible Gas Control System, capability will be provided to control gas concentrations in the containment by purging the containment through the Standby Gas Treatment System.

i) Contribution of combustible gases from secondary sources such as decomposition of coating or corrosion will be included in sizing the Combustible Gas Control System and backup purge system.

960110

j) Regulatory Guide 1.7 indicates that the hydrogen concentration limit is four percent by volume if more than five percent by volume oxygen is present. Since the Containment will not be inerted (oxygen - deficient) the oxygen concentration will always be above 5 percent by volume; hence, hydrogen concentration will be the control limit. Control measures will be taken to assure that the hydrogen concentration will not exceed four percent by volume in either the drywell or the Containment. 37(C)

components are shown in Table 6.2-25.

The purge fan discharges into the recirculation branch of the SGTS at a point downstream of the check valve, preventing an unfiltered discharge of drywell purge air to the outside.

Radioactive materials in the purge discharge airstream are in this way directed into the Shield Building annulus where mixing and holdup occur prior to treatment by the SGTS.

6.2.5.3 Design Evaluation

In evaluating the Combustible Gas Control System design, it was found to be necessary to make calculations regarding:

a) Hydrogen generated in the post-LOCA environment.

b) Resultant drywell and Containment concentrations.

c) The consequent functional requirements of the Combustible Gas Control System

6.2.5.3.1 Sources of Hydrogen

6.2.5.3.1.1 Short Term Hydrogen Generation

In the period of immediately after the LOCA, hydrogen would be generated by both radiolysis and metal-water reaction. However, in evaluating short-term hydrogen generation, the contribution from radiolysis is insignificant in comparison with the hydrogen generated by a postulated one percent metal-water reaction.

The generation of hydrogen by metal-water reaction is dependent upon the temperature of the cladding and the time at temperature. Based on loss of coolant calculational procedures established by the AEC in the "Interim Acceptance Criteria for Emergency Core Cooling System," the extent of metal-water reaction in the BWR/6 core is negligible (see Section 6.3 of the PSAR and GE Topical Report NEDO-10569, 11013-77 and NEDO 10723). The design of the BWR/6 ECCS is such that the real zirconium temperature is 1500 F; at this temperature, virtually no metal-water reaction occurs, and therefore hydrogen production by this means is insignificant.

Figure 6.2-28a shows drywell hydrogen concentration as a function of time based on the assumptions listed below. Based on these assumptions, it is calculated that the control limit (4%) would be reached in the drywell approximately 8 hours after the LOCA. 37(G) 53(U)

The transient hydrogen concentration curve shown in Figure 6.2-28a is based upon a non-condensible air pressure of 14.7 psia and a drywell air volume reduced by water flooding up to the top of the weir wall. This quasi steady state condition is conservatively assumed to exist immediately following the hypothesized metal-water reaction at LOCA + 2 minutes. This takes no credit for steam purging of the drywell or steam dilution due to pressure increase in the drywell. In a more realistic sequence, the 37(G)

960112

6.2.5.3.2 Analysis

The loss-of-coolant accident is described in Section 6.3. As previously noted, the occurrence of a one percent metal-water reaction cannot be related to any credible physical process during a LOCA. Therefore the following assumptions were made with respect to the sequence of events occurring after a LOCA. 28

a) As mentioned above, it is assumed that the metal-water reaction begins immediately after the LOCA and proceeds for two minutes at a rate that results in hydrogen generation equal to the larger amount determined by one of the two methods previously mentioned. 53(U)
In this case the second method described (five times the calculated value) results in the larger value (.85% vs .67%).

b) Radiolysis begins immediately and proceeds at a rate consistent with the assumptions stated in Section 6.2.5.3.1.2.

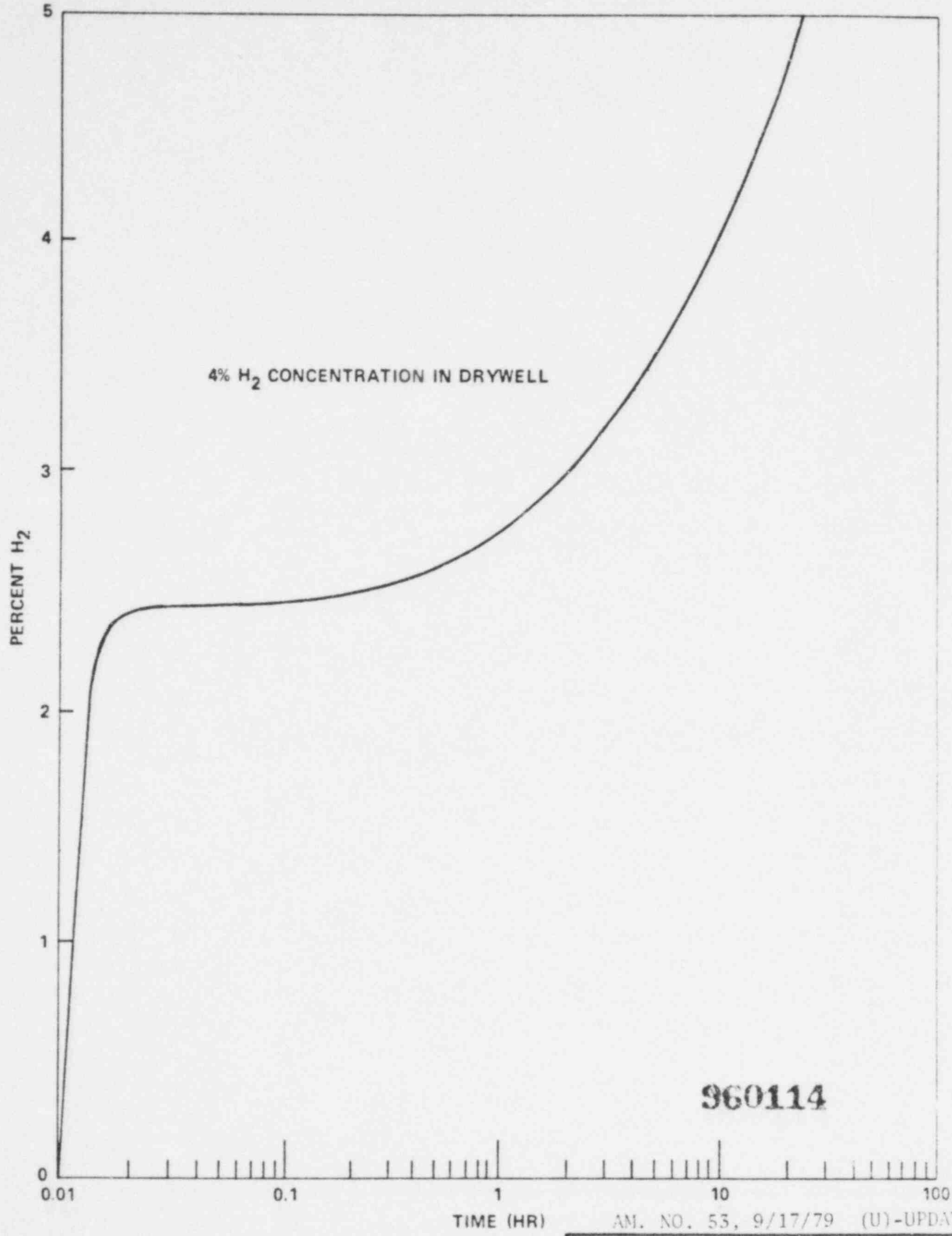
Based on these assumptions, it is calculated that the hydrogen concentration in the drywell will reach four volume percent approximately eight hours after LOCA. 46(C)
To calculate the redistribution of hydrogen between drywell and Containment due to Hydrogen Mixing System operation, a computer model of the system is used. The model predicts the volume concentration of each atmospheric constituent in each region as a function of time.

The calculations indicate that actuation of the Drywell Hydrogen Control System causes an immediate decrease in the drywell hydrogen concentration. (See Figure 6.2-29). The recirculation of the Containment atmosphere causes the hydrogen concentration to become uniform throughout the Containment and drywell. 26

Eventually, radiolytic generation may cause the hydrogen to again approach four volume percent. If this were to occur the Hydrogen Recombiner Subsystem would be manually activated. This system is designed to keep the hydrogen concentration below four volume percent. 37(G)

Both the mixing system and the recombiner subsystem are manually activated systems. 26
However, the mixing system would not be activated until the new hydrogen monitoring instrumentation indicates that the four percent limit of Regulatory Guide 1.7 is being approached. 26
A realistic analysis indicates that the ECCS would operate to limit the amount of metal-water reaction, and therefore, the amount of metal-water reaction generated hydrogen, to undetectable amounts. ECCS operation will also prevent fuel rod perforations; thus, no significant fission products would be released, and there would be no significant radiolysis in the pressure Suppression Pool. The only significant source of hydrogen, then, would be radiolysis of the coolant in the core region. Based on this realistic analysis, no action to control hydrogen buildup would be required until weeks after the LOCA. 37(G)

960113



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AM. NO. 53, 9/17/79 (U)-UPDATE

HOUSTON LIGHTING & POWER COMPANY
Allens Creek Nuclear Generating Station
Unit 1

HYDROGEN CONCENTRATION IN THE DRYWELL (WITHOUT BLOWER ACTIVATION .85% METAL-WATER REACTION

FIGURE 6.2-28a

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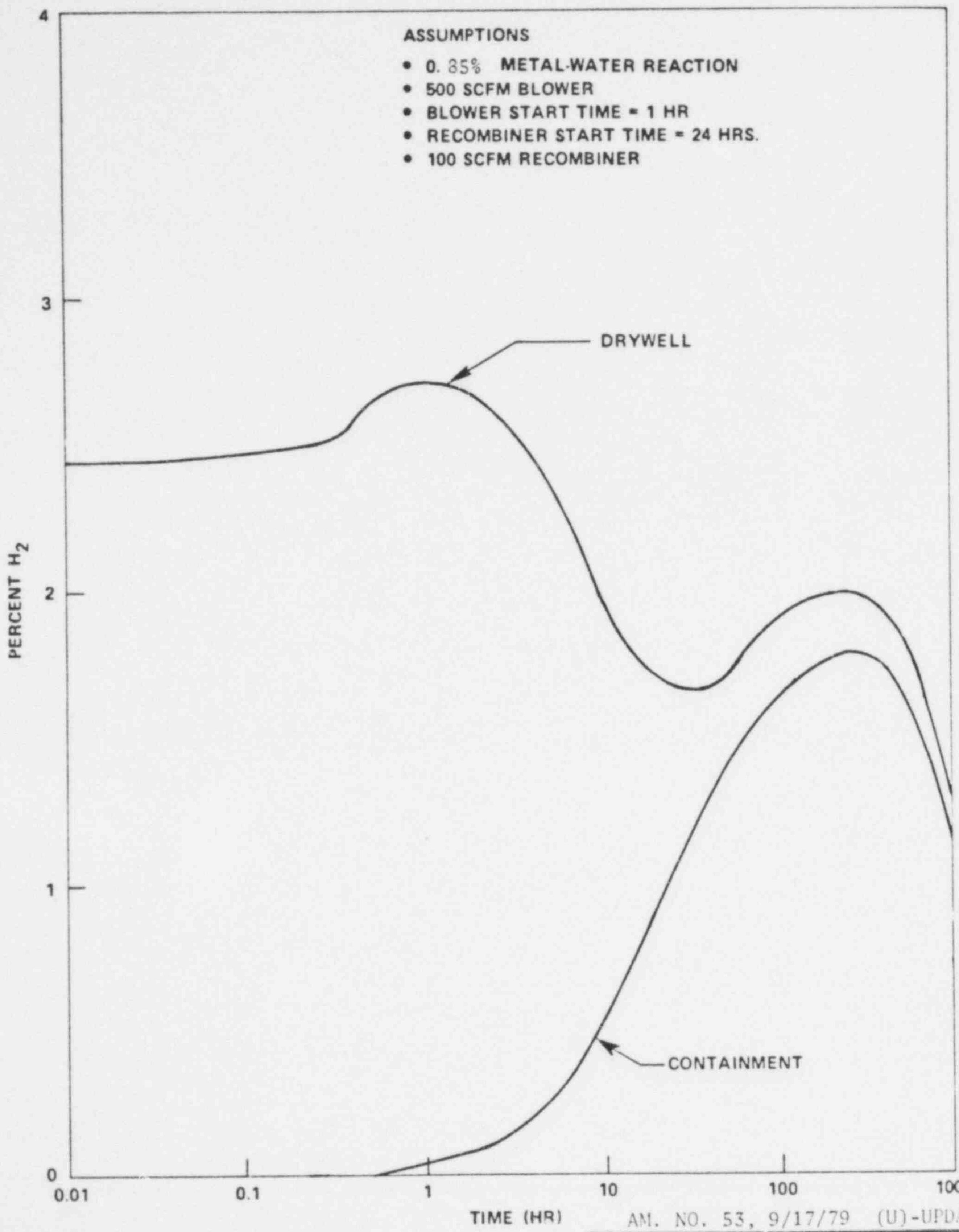

AM. NO. 53, 9/17/79 (U)-UPDATE

HOUSTON LIGHTING & POWER COMPANY
Allens Creek Nuclear Generating Station
Unit 1

HYDROGEN CONCENTRATION IN THE DRYWELL (FOLLOWING BLOWER ACTIVATION)

FIGURE 6.2-29

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Chapter 9

AUXILIARY SYSTEMS

EFFECTIVE FIGURES LISTING

Chapter 9

AUXILIARY SYSTEMS

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CHAPTER 9

AUXILIARY SYSTEMS

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9.1.2 SPENT FUEL STORAGE

9.1.2.1 Design Bases

9.1.2.1.1 Safety Design Bases

a) The fuel array in the fully loaded spent fuel racks shall be subcritical with a k_{eff} of less than or equal to 0.95 for all conditions. 53(D)

b) Each spent fuel storage rack containing its full complement of fuel will be designed to withstand "specified loads" to minimize distortion of the fuel storage arrangement.

c) Shielding for the spent fuel storage arrangement will be sufficient to protect plant personnel from exposure to radiation in excess of published guideline values.

d) The spent fuel storage facility will be designed to prevent missiles generated by high winds from causing damage to the fuel.

e) The spent fuel storage facility ventilation system will be designed to limit the potential offsite exposures in the event of significant release of radioactivity from the fuel as detected by a high radiation signal. The signal will shutdown normal ventilation systems and automatically place in operation the Standby Gas Treatment System, filtering all exhaust air prior to its release to the environment. 1 Q9.3

f) The spent fuel storage racks will be designed to seismic Category I requirements.

9.1.2.1.2 Power Generation Design Bases

a) Spent fuel storage space will be supplied to initially accomodate 1710 fuel bundles in high-density spent fuel storage racks (5 years normal batch discharges plus a full core reserve), with the capability to add high-density storage racks for up to 2790 fuel bundles (10 years normal batch discharges plus a full core reserve). 53(D)

b) Spent fuel storage racks will be designed and arranged so that the fuel assemblies can be handled efficiently during refueling operations.

9.1.2.2 Facilities Description

Spent fuel storage racks will provide a place in the Fuel Pool for storing spent fuel received from the reactor vessel. These will be top-entry racks designed to maintain the spent fuel in a space geometry that precludes the possibility of criticality under all conditions. 53(D)

The loaction of the spent fuel storage facilities within the station complex is shown in Figure 1.2-17a. Spent fuel can be stored both in the Reactor Building and in the Fuel Handling Building. However, fuel will not 37(D)

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be stored in the Reactor Building except during periods of refueling, on a temporary basis. The Containment Fuel Pool will have a capacity for storing 25 percent of a core and spent fuel storage in the FHB can accommodate 1710 fuel bundles. (with the capability to expand to 2790 fuel bundles) 37(D) 53(D)

The rack arrangement is designed to prevent accidental insertion of fuel bundles between adjacent racks. The storage rack structure is so designed that the upper tie plate casting cannot be lowered below the top of the upper rack. This prevents any tendency of the fuel bundles jamming on insertion or removal from the rack. The rack spacing is such as to maintain minimum spacing of adjacent racks for geometric reactivity control. 37(D)

9.1.2.3 Safety Evaluation

The design of the FHB high-density spent fuel storage racks will provide for a subcritical multiplication factor (k_{eff}) of 0.95 or less for both normal or abnormal storage conditions. Normal conditions exist when the fuel storage racks are located in the pool and are covered with a normal depth of water (about 25 feet above the stored fuel) for radiation shielding and with the maximum number of fuel assemblies or bundles in their design storage position. The spent fuel will be covered with water at all times by a minimum depth required to provide sufficient shielding. An abnormal condition may result from accidental dropping of equipment without first disengaging the fuel from the hoisting equipment. To ensure that the design criteria are met, the following normal and abnormal spent fuel storage conditions are analyzed: 53 (D) 53(D)

a) Closest eccentric fuel positioning in the spent fuel storage array

b) Pool water temperature increases to 212^{o} F.

c) Moving fuel bundle in regions peripheral to the storage rack array 53 (D)

d) Fuel bundle drop on or through a rack

Objects such as fuel handling fixtures that could conceivably fall into the fuel would not transfer energy amounts exceeding the specified limits of the fuel racks. The design of the Fuel Handling Building does not require that the spent fuel shipping cask be lifted to the operating floor level. This design precludes any movement of the cask over the storage pools.

Any connections to the Spent Fuel Pool will be designed such that the watertight integrity of the pool is not compromised (See Section 9.1.3.3).

An appropriate air handling system with radiation sensing instrumentation in the exhaust ducts will be provided to adequately control radiation leakage to the environment (See Section 9.4.5).

In case of a significantly high radiation level, the exhaust ducts will close, and the Standby Gas Treatment System (Section 6.2.3) will be activated.

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For the foregoing analysis, it is concluded that the spent fuel storage arrangement meets its design bases and satisfies the requirements of Regulatory Guide 1.13.

Offsite exposure to release of radioactive products from damaged or failed fuel in the fuel building is dependent on three systems. The functions of these systems as described in Section 9.4.5 are:

a) The ventilation exhaust radiation monitoring system detects radioactivity in the fuel building atmosphere

b) The Standby Gas Treatment System minimizes the release of contaminated air to the environment

c) The fuel building isolation control system automatically closes isolation dampers to block potential leakage of contaminated air to the environment

d) The fuel storage facilities will be designed to seismic Category I requirements to prevent earthquake damage to the stored fuel.

From the foregoing analyses, it is concluded that the spent fuel storage arrangement and design meet the safety design bases and satisfy the intent of Regulatory Guide 1.13.

The spent fuel storage racks will be designed to meet seismic Category I requirements (See Section 3.2). Stresses in a fully loaded rack will not exceed stresses specified by the ASME B&PV Code, Section III, Subsection NF-3400. 53(D)

The spent fuel storage racks will be made from type 304 stainless steel. Materials used for construction will be specified in accordance with the latest issue of applicable ASTM specifications (A-240, A-479).

Control of weights handled within the fuel building will be administrative control of crane movement and location of spent fuel bundles in the fuel storage pool. The 15 ton general purpose crane will traverse the full length of the fuel building. 3 Q9.1 Q9.2

Administrative controls will specify the fuel storage pool area to be filled with spent fuel bundles for decay storage, in order to assure that if required, the fuel storage pool could be traversed by the loaded 15 ton general purpose building crane without moving a load over stored spent fuel.

Handling operations involving the 15 ton general purpose building crane will avoid the area of the new fuel vault by moving loads over the cask pool. 37(C)

Transfer of fuel assemblies between the new fuel vault and the water filled storage pool and also within the storage pool, transfer pool and cask vault is performed with the fuel handling platform. The fuel grapple or the auxiliary fuel hoist is used depending on the transfer operation. The grapple and hoist provided with load sensing and limiting devices designed to the following load limits:

(C) Consistency
(D) Design

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	Fuel Grapple (lbs)	Auxiliary Hoist (lbs)
load limiting switch	1200	1000
load sensing switch	485	485
stall torque of hoist system	3000	3000

3
Q9.2

The load limiting features of the platform fuel grapple and auxiliary hoist will prevent damage to the fuel racks during fuel transfer operations.

These load limits provide a redundant safety feature since the fuel handling grapples are not lowered below the upper fuel rack and they will interface with the fuel bail, only.

9.1.2.4 Testing and Inspection

The spent fuel storage racks require no periodic special testing or inspection for nuclear safety purposes.

9.1.2.5 Summary of Radiological Considerations

By adequate design and careful operation procedures, the safety design bases of the spent fuel storage arrangement are satisfied. Thus, the exposure of plant personnel to radiation is maintained well below published guideline values. Further details of radiological considerations including those for the spent fuel storage arrangement are presented in Chapter 12.

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9.1.3 SPENT FUEL POOL COOLING AND CLEANUP SYSTEM

9.1.3.1 Design Bases

The Spent Fuel Pool Cooling System will be Safety Class 3, seismic Category I in compliance with Regulatory Guides 1.13 and 1.26. The cleanup portion of the Spent Fuel Pool Cooling and Cleanup System will be Safety Class "other" and "non-seismic", and can be isolated with redundant isolation valves which are part of the safety related fuel pool cooling system. 23

The objective of the Spent Fuel Pool Cooling and Cleanup System will be to (1) remove the decay heat from the fuel assemblies, (2) control water clarity, (3) maintain upper pool water temperature and clarity to permit normal reactor refueling and servicing and maintain fuel pool water level such that fuel is not uncovered. 37(U)

The Spent Fuel Pool Cooling and Cleanup System will be designed to:

a) Minimize corrosion product buildup and control water clarity, so that the fuel assemblies can be efficiently handled underwater.

b) Minimize fission product concentration in the water which could be released from the pool to the fuel handling area environment.

c) Monitor Spent Fuel Pool and containment fuel transfer pool water level and maintain a water level above the fuel sufficient to provide shielding for normal building occupancy.

d) Maintain the pool water temperature below 125 F under normal operating conditions. The maximum normal heat load in the spent fuel pool consists of one quarter of the reactor core fuel assemblies having decayed 80 hours (the minimum time required to transfer spent fuel from the reactor to the storage pool) and plus successive annual batch discharges (which have also been irradiated for 4 full power years) and have cooled for periods of 1-9 years. 37(D) 53(D)

e) Remove drywell heat transferred to the containment pool above the drywell. 37(D)

f) Cooling portion will be Safety Class 3, seismic Category I, and cleanup portion will be Safety Class "other" and "non-seismic". 23

The reactor will be loaded with a fuel charge of 732 fuel bundles. The Spent Fuel Pool storage capacity in the Fuel Handling Building will be 1710 fuel bundles. (with the capability to expand to 2790 fuel bundles) 37(D) 53(D)

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The heat sources are based on full power operation for 4 years prior to removal of fuel assemblies from the reactor using the methodology of USNRC 53(D)

(U) Update
(D) Design

Standard Review Plan Branch Technical Position APCSB 9-2 and the fission product decay heat rates of ANS Standard 5.1, October 1973. 53 (D)

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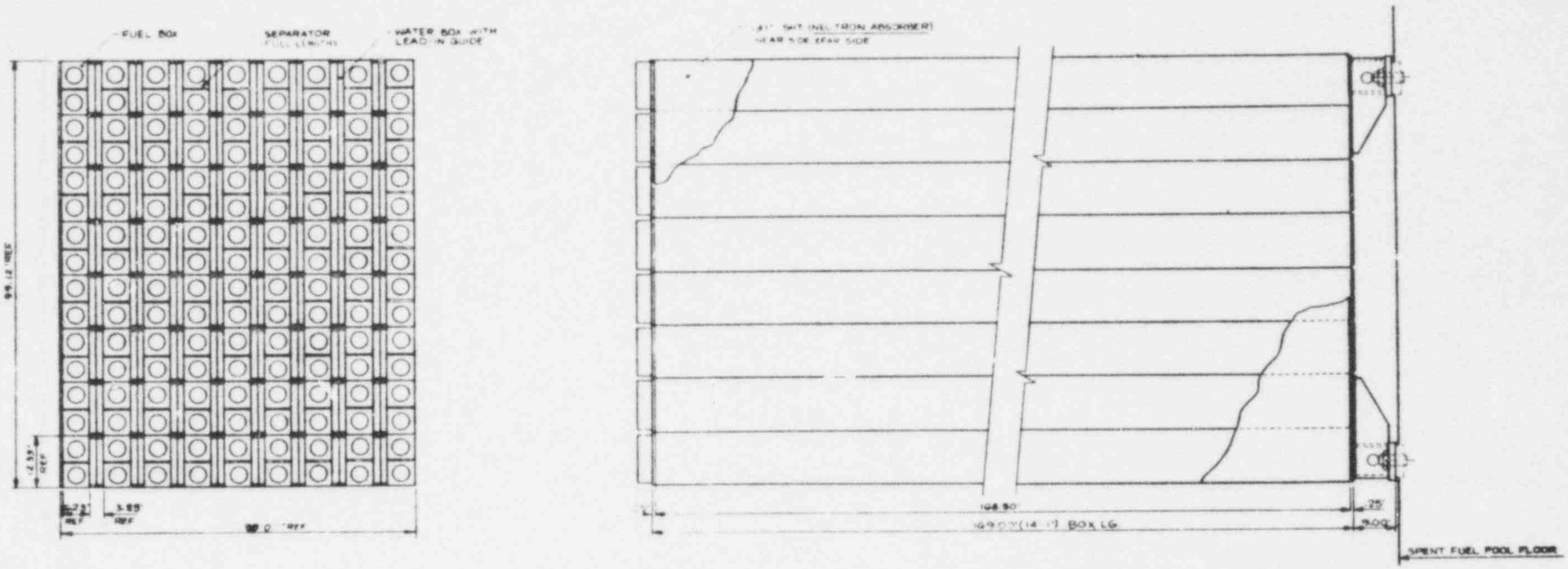
FIGURE 9.1-2



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Houston Lighting & Power Company
Allens Creek Nuclear Generating Station
Unit 1

High Density Spent Fuel Storage Racks
FIGURE 9.1-2a

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HOUSTON LIGHTING & POWER COMPANY
Allens Creek Nuclear Generating Station
Units 1 & 2

FIGURE 9.1-12a

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HOUSTON LIGHTING & POWER COMPANY
Allens Creek Nuclear Generating Station
Units 1 & 2

FIGURE 9.1-12b

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HOUSTON LIGHTING & POWER COMPANY
Allens Creek Nuclear Generating Station
Units 1 & 2

FIGURE 9.1-12C