UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

August 6, 1992

Docket Nos. 50-498
and 50-499

LICENSEE: Houston Lighting and Power Company

FACILITY: South Texas Project, Units 1 and 2

SUBJECT: SUMMARY OF MEETING HELD ON JULY 23, 1992 TO DISCUSS PROPOSED TECHNICAL SPECIFICATION AMENDMENTS RELATED TO ALLOWED OUTAGE TIMES AND SURVEILLANCE TEST INTERVALS (TAC NOS. M76048 AND M76049)

On July 23, 1992, members of the NRC staff and a contractor from Brookhaven National Laboratories (BNL) met with representatives of Houston Lighting and Power Company to discuss proposed Technical Specification (TS) amendments related to allowed outage times (AOTs) and surveillance test intervals (STIs). A list of the attendees is provided as Enclosure 1. The slides used by the licensee during the meeting are provided as Enclosure 2.

The licensee's submittal of February 1, 1990, proposed to change the AOTs and STIs associated with several systems using the plant specific probabilistic safety assessment (PSA) as the primary justification for the revisions. A subsequent transmittal dated November 27, 1990, withdrew the proposed changes associated with those systems which were determined to be the most significant in terms of an increased risk of core damage resulting from an extension of the related system AOTs or STIs. In addition, several of the other proposed changes included in the initial submittal were superceded by subsequent licensee amendment requests. The NRC review of the internal events and fire protection portions of the plant specific PSA was documented in a safety evaluation which was issued on January 21, 1992.

Discussions during the meeting provided the staff and contractor with background information regarding the PSA and the analyses performed to justify the proposed Technical Specification changes. In response to questions regarding the incorporation of plant specific performance data into the PSA models, the licensee stated that, at this time, only the Essential Cooling Water (ESW) data has been compiled, incorporated, and verified to be conservative with respect to initial PSA assumptions. Plant specific performance data for other systems will be incorporated into the future updates of the PSA model assumptions. Several other questions and proposed actions were discussed and will be addressed by a formal request for additional information to be issued by the NRC.

NRC staff indicated their plans to obtain the Riskman computer code from Pickard, Lowe and Garrick, Inc. The staff expressed a desire to obtain the STP specific models after the initial familiarization with the Riskman code and related methodologies. The licensee stated that the possibility of supplying the STP models would be investigated as well as any necessary

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constraints on the use of the models if they were provided. A conference call will be held in August or September 1992 to discuss the NRC's progress related to the acquisition of Riskman and the possible use of the STP models.

In a telephone conversation held subsequent to the meeting, the licensee proposed that the TSs related to the Reactor Protection System and Engineered Safety Features Actuation System changes be reviewed as soon as possible in order to support the facilities' trip reduction program. This request will be formally documented by a letter from the licensee and the NRC has given preliminary agreement to proceed with the separate review. The proposed changes to Reactor Protection System and Engineered Safety Features Actuation System specifications are similar to those approved for other facilities based upon analyses performed by Westinghouse.

Original Signed By

William D. Reckley, Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. List of Attendees
2. Licensee Presentation

cc w/enclosure:
See next page

DISTRIBUTION:

Docket File	WReckley
NRC/PDR	GDick
Local PDR	EPeyton
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NAME	EPeyton	WReckley;nb	SBlack		
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cc w/enclosure:
Mr. J. Tapia
Senior Resident Inspector
U.S. Nuclear Regulatory Commission
P. O. Box 910
Bay City, Texas 77414

Mr. J. C. Lanier/M. B. Lee
City of Austin
Electric Utility Department
P. O. Box 1088
Austin, Texas 78767

Mr. K. J. Fiedler
Mr. M. T. Hardt
City Public Service Board
P. O. Box 1771
San Antonio, Texas 78296

Mr. D. E. Ward
Mr. T. M. Puckett
Central Power and Light Company
P. O. Box 2121
Corpus Christi, Texas 78403

INPO
Records Center
1100 Circle 75 Parkway
Atlanta, Georgia 30339-3064

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 1000
Arlington, Texas 76011

Mr. Joseph M. Hendrie
50 Bellport Lane
Bellport, New York 11713

Judge, Matagorda County
Matagorda County Courthouse
1700 Seventh Street
Bay City, Texas 77414

Mr. William J. Jump
Manager, Nuclear Licensing
Houston Lighting and Power Company
Box 289
th, Texas 77483

Jack R. Newman, Esq.
Newman & Holtzinger, P.C.
1615 L Street, N.W.
Washington, D.C. 20036

Licensing Representative
Houston Lighting and Power Company
Suite 610
Three Metro Center
Bethesda, Maryland 20814

Bureau of Radiation Control
State of Texas
1101 West 49th Street
Austin, Texas 78756

Rufus S. Scott
Associate General Counsel
Houston Lighting and Power Company
P. O. Box 618
Houston, Texas 77208

Mr. Donald P. Hall
Group Vice-President, Nuclear
Houston Lighting & Power Company
Houston, Texas 77251

ENCLOSURE 1

List of Attendees

Houston Lighting and Power

Richard Murphy
Rick Grantom
Sam Phillips

NRC

Erulappa Chelliah
Millard Wohl
William Reckley
Robert Evans

Brookhaven National Laboratories

Pranab Samanta

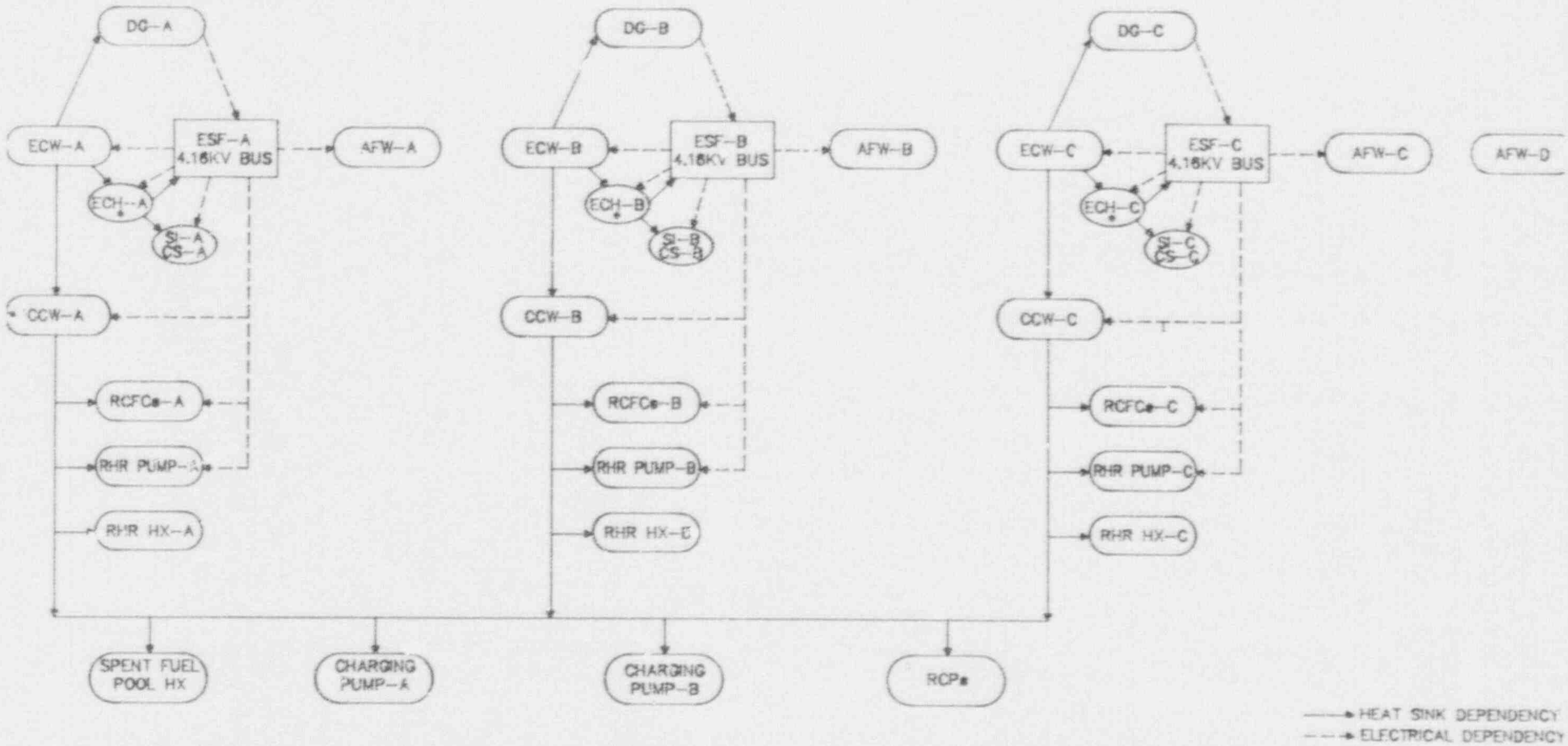
THREE TRAIN DESIGN

SOUTH TEXAS PROJECT
ELECTRIC GENERATING STATION

SOUTH TEXAS PROJECT

- Three-train ESF systems
  - Physically segregated
  - Electrically independent

THREE TRAIN DESIGN HEAT SINK AND ELECTRICAL DEPENDENCY



* NOTE: AN ECW PUMP TRAIN CAN SERVE AS A HEAT SINK FOR TWO ECH TRAINS.

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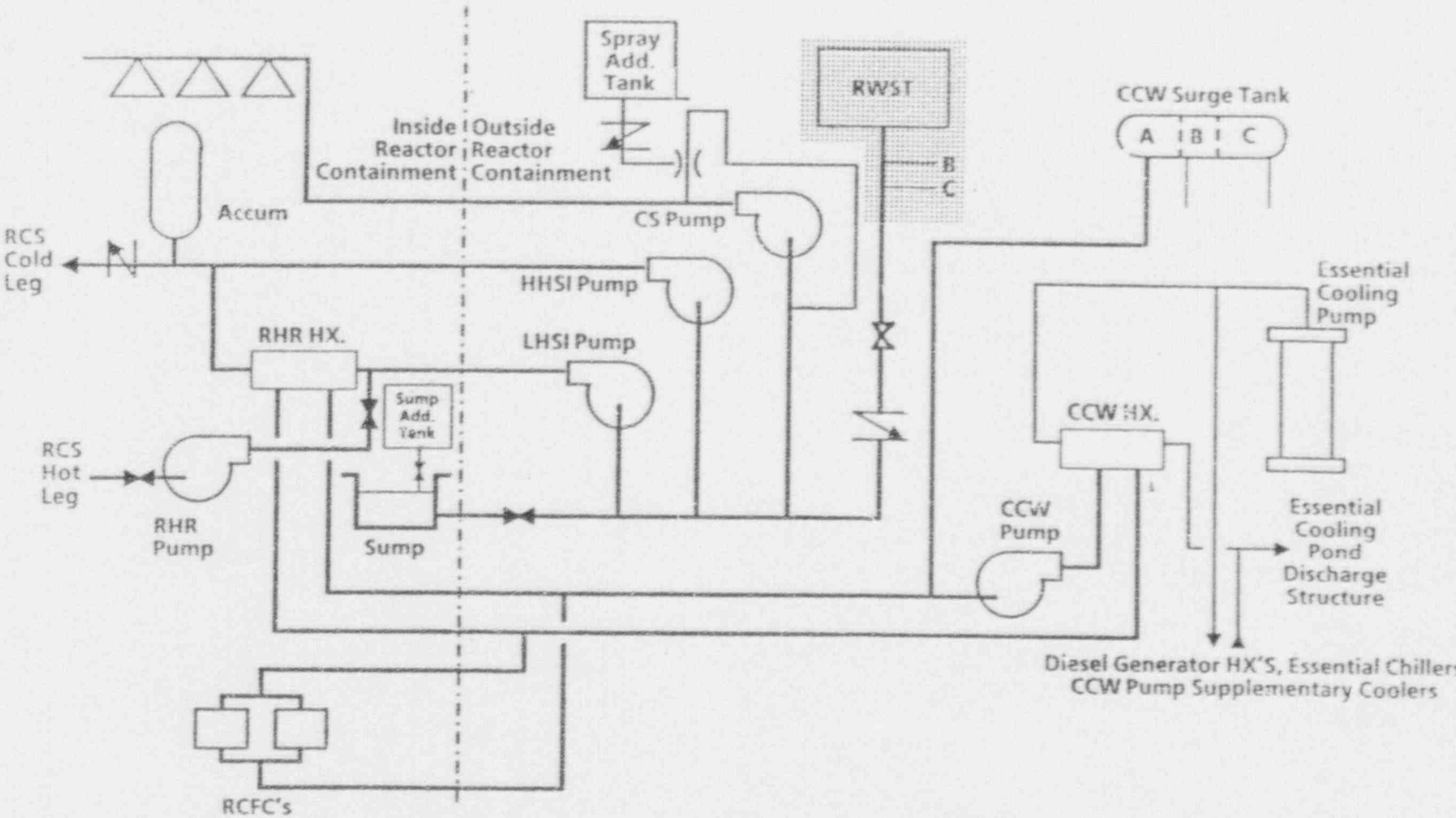
PM Outages Typical for Unit 1 Systems

Week	Sun.	Mon.	Tue.	Wed.	Thr.	Fri.	Sat.
1		CCW-A			ECW-A		
2							
3			AFW-C				
4							
5			AFW-A				
6		CCW-B			ECW-B		
7							
8			AFW-D				
9							
10			AFW-B				
11		CCW-C			ECW-C		
12							
1		CCW-A			ECW-A		



INTERRELATIONSHIP OF ESF SYSTEMS IN THE THREE-TRAIN CONCEPT
(Typical of Three)

STP

SOUTH TEXAS PROJECT THREE TRAIN SYSTEMS

System	Trains Installed	Minimum number of trains required		
		Normal Operation / Shutdown	Large Breaks	Other Accidents
Diesel Generators	3	0/0	2	1
Essential Cooling Water	3	1/1	2	1
Component Cooling Water	3	1/1	2	1
Reactor Containment Fan Coolers	3 (2 RCFC Units per train)	2/2*	2 (3 RCFC Units)	1
Safety Injection	3	0/0	2**	1
Containment Spray	3	0/0	2	0
Residual Heat Removal Heat Exchangers	3	0/1	2	1
Auxiliary Feedwater	4	0/0	2	1

* Normally supplied by RCB chilled water
** Share RHR exhanger - RHR pumps not required

CORE DAMAGE FREQUENCY

Description	Core Damage Frequency
Total Internal Initiating Events	1.7×10^{-4}
Total External Initiating Events	1.2×10^{-6}
Total Core Damage Frequency	1.7×10^{-4}

Table 5.2-1 (Page 1 of 2). Initiating Event Categories Selected for Quantification of the South Texas Project Risk Model		
Group	**Initiating Event Categories Selected for Separate Quantification**	**Code Designator**
Loss of Coolant Inventory	1. Excessive LOCA	ELOCA
	2. Large LOCA	LLOCA
	3. Medium LOCA	MLOCA
	4. Small LOCA	
	a. Nonisolable	SLOCA
	b. Isolable	ILOCA
	5. Interfacing Systems LOCA	V
	6. Steam Generator Tube Rupture	SGTR
Transients	7. Reactor Trip	RT
	8. Turbine Trip	TT
	9. Loss of Condenser Vacuum	LCV
	10. Closure of All MSIVs	AMSIV
	11. Steam Line Break Upstream of MSIVs	
	a. Steam Line Break inside the Containment	SLBI
	b. Main Steam Relief or Safety Valve Opening	MSV
	12. Steam Line Break Downstream of MSIVs	SLBD
	13. Inadvertent Safety Injection	SI
	14. Miscellaneous Transients	
	a. Total Main Feedwater Loss (includes feedwater line break)	TLMFW
	b. Partial Main Feedwater Loss	PLMFW
	c. Excessive Feedwater	EXFW
	d. Closure of One Main Steam Isolation Valve	1MSIV
	e. Core Power Excursion	CPEXC
	f. Loss of Primary Flow	LOPF
Common Cause Initiating Events		
Support System Faults	15. Loss of Offsite Power	
	a. Events External to Plant	LOOP
	b. Transient-Induced Loss of Offsite Power	TLOOP
	16. Loss of One DC Bus	
	a. Loss of DC Bus E1A11	L1DCA
	b. Loss of DC Bus E1B11	L1DCB

Table 5.2-1 (Page 2 of 2). Initiating Event Categories Selected for Quantification of the South Texas Project Risk Model

Group	Initiating Event Categories Selected for Separate Quantification	Code Designator
	17. Total Loss of Essential Cooling Water	LOECW
	18. Total Loss of Component Cooling Water	LOCCW
	19. Loss of Instrument Air	LOIA
	20. Loss of EAB HVAC	LOEAB
	21. Loss of Control Room HVAC	LOCR
Seismic Events	22. 0.1g Seismic	SEIS1
	23. 0.2g Seismic	SEIS2
	24. 0.4g Seismic	SEIS4
	25. 0.6g Seismic	SEIS6
Fires	26. Control Room — Loss of All Three Motor-Driven AFW Pumps	FIRE10
	27. Control Room — Loss of Control Room HVAC and EAB HVAC	FIRE18
	28. Control Room — Loss of All AFW Pump Trains	FIRE23
Flooding (External)	29. Loss of Offsite Power and PD Pump	FLOOD1
	30. Loss of Offsite Power, PD Pump, and All Three Emergency Diesel Generators	FLOOD2
	31. Loss of Offsite Power, PD Pump, and Loss of All ECW	FLOOD3
	32. Loss of Offsite Power, PD Pump, and Loss of All CCW	FLOOD4
	33. Loss of Offsite Power, PD Pump, All CCW, and Train C Essential Chillers	FLOOD5
	34. Loss of Offsite Power, PD Pump, and Train/B RCFCs	FLOOD6
	35. Loss of Offsite Power, PD Pump, Train B, AC Power, and Main Control Room	FLOOD7
Flooding (Internal)	None	
Others	36. Severe Wind (tornado) — Loss of Offsite Power, Loss of TSC Diesel Generator, and Plugging of ECW Pump Travelling Screen by Debris	FLECW

STPEGS Frontline and Support Systems Analyzed in Detail

- Support Systems
  - Essential Chilled Water System
  - Electric Power (including Diesel Generators)
  - Component Cooling Water System
  - Essential Cooling Water System
  - Reactor Trip System
  - Solid State Protection System
  - Engineered Safety Features Actuation System
  - Qualified Display Processing System
  - Electrical Auxiliary Building HVAC System
  - Control Room HVAC System

STPEGS Frontline and Support Systems Analyzed in Detail

- o Frontline Systems
  - Reactor Containment Fan Coolers
  - Containment Isolation System
  - Emergency Core Cooling System
    - o Refueling Water Storage Tank and Suction Lines
    - o Containment Pump Suction Lines
    - o High Head Safety Injection
    - o Low Head Safety Injection
  - Auxiliary Feedwater System and Secondary Steam Relief
  - Containment Spray System
  - Primary Pressure Relief
  - Main Steam Isolation Systems
  - Chemical and volume Control
  - Residual Heat Removal

IMPACT OF TECH SPECS ON PLANT AVAILABILITY

- o LIMITED CREDIT IS GIVEN IN THE CURRENT STPEGS TECH SPECS FOR THE THREE TRAIN DESIGN AND THE ADDITIONAL SAFETY EQUIPMENT. WHEN A PUMP, FOR EXAMPLE, IS DECLARED INOPERABLE IN A "TWO TRAIN" PLANT ONLY ONE REMAINS AVAILABLE, WHEREAS AT STPEGS TWO MUST REMAIN AVAILABLE. THE PLANT CAN BE BROUGHT TO A STABLE CONDITION WITH ONE TRAIN AVAILABLE EXCEPT IN THE CASE OF THE LOW FREQUENCY DESIGN BASIS LARGE BREAK LOCA.

- o LCOs CONTRIBUTE TO PLANT UNAVAILABILITY.

- o THE MORE EQUIPMENT UNDER TECH SPECS, THE HIGHER THE PROBABILITY OF ENTERING AN LCO.

- o THE MORE LCOs A PLANT ENTERS, THE GREATER THE CHANCE FOR SHUTTING THE PLANT DOWN AND INCREASING PLANT UNAVAILABILITY.

- o THE STPEGS TECH SPECS REQUIRE TWO OUT OF THE THREE ESF TRAINS TO BE AVAILABLE BASED ON A LOW FREQUENCY POSTULATED DESIGN BASIS LARGE BREAK LOCA. IN THIS CASE, ONE TRAIN OF EQUIPMENT WILL BE EFFECTIVE EXCEPT IN THE INSTANCE IN WHICH THE AVAILABLE TRAIN FEEDS THE BROKEN LOOP, WHICH IS VERY UNLIKELY. STPEGS IS MORE LIKELY TO ENTER AN LCO DUE TO THE ADDITIONAL EQUIPMENT THAT MUST BE AVAILABLE.

- o AS A RESULT, STPEGS MAY EXPERIENCE A PLANT AVAILABILITY SOMEWHAT LESS THAN THE US AVERAGE BECAUSE IT HAS MORE SAFETY EQUIPMENT.

Table 2-1

List of Technical Specifications Evaluated

Technical Specification	System	Current AOT (Days)	Current STI (Days)	Proposed AOT (Days)	Proposed STI (Days)
3.1.2.4	Chemical and Volume Control (i.e. Charging Pumps)	3	N/C	10	N/C
4.3.1	Reactor Protection	N/C	62	N/C	92
4.3.2	Engineered Safeguard Features Actuation	N/C	62	N/C	92
3.4.2.2	Pressurizer Safety Valves	15 min	N/C	1 Hour	N/C
3.4.4	Pressurizer PORVs	1 Hour	N/C	6 Hours	N/C
3.5.1	Accumulators	1 Hour	N/C	12 Hours	N/C
3.5.2	Emergency Core Cooling	3	N/C	10	N/C
3/4.5.6	Residual Heat Removal	3	92	10	184
4.6.1.7	Containment Ventilation	N/C	31	N/C	92
3/4.6.2.1	Containment Spray	3	92	10	184
3.6.2.2	Containment Spray Additive	3	N/C	10	N/C
3/4.6.2.3	Reactor Containment Fan Coolers	3	31	10	92
3.6.3	Containment Isolation	4 Hours	N/C	24 Hours	N/C
3.7.1.1	Steam Generator Safety Relief Valves	4 Hours	N/C	24 Hours	N/C
3/4.7.1.2	Auxiliary Feedwater	3	31	10	92
3.7.3	Component Cooling Water	3	N/C	10	N/C

Table 2-1 (Cont'd)

List of Technical Specifications Evaluated

Technical Specification	System	Current AOT (Days)	Current STI (Days)	Proposed AOT (Days)	Proposed STI (Days)
3.7.4	Essential Cooling Water	3	N/C	10	N/C
3/4.7.7	Control Room HVAC	(1)	31	(2)	92
4.7.13	Electrical Auxiliary Building HVAC	N/C	12 Hours	N/C	24 Hours
3.7.14	Essential Chilled Water	3	N/C	10	N/C
3.8.1.1	Diesel Generators	(3)	N/C	(4)	N/C
3.8.2	DC Electrical Sources	(5)	N/C	(6)	N/C

NOTES:

N/C No Change

(1) 7 days for the first inoperable train of control room HVAC and 24 hours for the second train of three.

(2) 10 days for the first inoperable train of control room HVAC and 72 hours for the second train of three.

(3) 72 hours for the first inoperable standby diesel generator and 2 hours for the second diesel generator.

(4) 10 days for the first inoperable standby diesel generator and 12 hours for the second diesel generator.

(5) 24 hours for Channels I and IV battery chargers; and 2 hours for any battery and Channels II and III chargers.

(6) 72 hours for any battery charger and 24 hours for any battery.

Tar e 1-1

Summary of Plant Level Results

Technical Specification	System	Change in Risk (%)
3.1.2.4	Chemical and Volume Control (i.e. Centrifugal Charging Pumps)	-0.7
4.3.1	Reactor Protection	0.2
4.3.2	Engineered Safety Features Actuation	0.0
3.4.2.2	Pressurizer Safety Valves	*
3.4.4	Pressurizer PORVs	*
3.5.1	Accumulators	0.3
3.5.2	Emergency Core Cooling	1.3
3/4.5.6	Residual Heat Removal	0.0
4.6.1.7	Containment Ventilation	*
3/4.6.2.1	Containment Spray	0.0
3.6.2.2	Containment Spray Additive	0.0
3/4.6.2.3	Reactor Containment Fan Coolers	-0.1
3.6.3	Containment Isolation	*
3.7.1.1	Steam Generator Safety Valves	*
3/4.7.1.2	Auxiliary Feedwater	30.4
3.7.3	Component Cooling Water	1.4 (+)
3.7.4	Essential Cooling Water	23.8 (+)
3/4.7.7	Control Room HVAC	0.5
4.7.13	Electrical Auxiliary Building HVAC	*
3.7.14	Essential Chilled Water	0.7 (+)
3.8.1.1	Diesel Generators	24.8
3.8.2	Single DC Power Train	1.6

* Denotes that a qualitative analysis is performed.
\+ Denotes that the AOT for all dependent systems are also increased accordingly and the integrated change in risk is shown.