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50-251 Turkey Point Plant, Unit 4, Florida Power and Light C 05000251
AUTH.NAME AUTHOR AFFILIATION
UHRIG,R.E. Florida Power & Light Co.
RECIP.NAME RECIPIENT AFFILIATION
EISENHUT,D.G. Division of Licensing

SUBJECT: Forwards responses to 810821 Questions 1,2 & 5 re pressurized thermal shack to reactor pressure vessels.

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	RES DE TI		1	1	RES DRA		1	1
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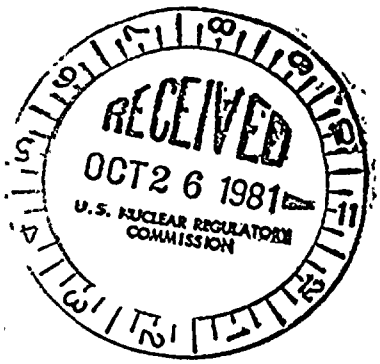
FLORIDA POWER & LIGHT COMPANY

October 21, 1981
L-81-462



Office of Nuclear Reactor Regulation
Attention: Mr. Darrell G. Eisenhut, Director
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Eisenhut:

Re: Turkey Point Units 3 & 4
Docket Nos. 50-250 and 50-251
Pressurized Thermal Shock to Reactor Pressure Vessels

Please find attached our response to Questions (1), (2) and (5) of your letter dated August 21, 1981. Our letter L-81-418 dated September 23, 1981 outlined our schedule for responding to the remainder of questions, and described our participation in the Westinghouse Owner's Group. The results of that effort may be used to supplement our response to Question (5) in the future.

Very truly yours,

J. a. de Mostry

Robert E. Uhrig
Vice President
Advanced Systems and Technology

REU/JEM/mbd

Attachment

cc: Mr. James P. O'Reilly, Region II
Mr. Harold F. Reis, Equire

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Pressurized Thermal Shock to Reactor Pressure Vessels

Question (1):

Provide the RT_{NDT} Values of the critical welds and plates (or forgings) in your vessel for:

(a) Initial (as built) conditions and location (e.g. 1/4T) and
(b) current conditions (include fluence level) at the RPV inside carbon steel surface.

Response (1):

		Material	Initial RT_{NDT}	ΔRT_{NDT}+	Current RT_{NDT}
(a)	Intermediate Inner Forging*	123P481VA1	+50 F	+ 35 F	+ 85F
	Circumferential (Girth) Weld**	SA 1101	+ 3 F	+190 F	+193F
	Lower Forging*	122S180VA1	+40 F	+ 35 F	+ 75F

* 1/4 T

** Inner wall. The current RT_{NDT} (1/4 T) = +168 F. Value is based on Unit 3 data which has been shown to be more representative of Unit 4 than surveillance capsule removed from Unit 4 (L-77-113, dated April 11, 1977 and L-77-326, dated October 21, 1977).

\+ Based on the slope of prediction curves presented in proposed ASTM Standards "Predicting Neutron Radiation Damage To Reactor Vessel Material."

(b) There have been 5.61 Effective Full Power Years (EFPY) of operation as of September 30, 1981 at Turkey Point Unit 4.

The total fluence on the inner wall is 1.1×10^{19} n/cm^2 and 6.6×10^{18} n/cm^2 at 1/4 T.

Question (2):

At what rate is RT_{NDT} increasing for these welds and plate material?

Response (2):

RT_{NDT} is increasing at the rate of 7°/F/EFPY for the next 10 years; for the remainder of life, 5°/F/EFPY. The rate of change for the forgings is 30% for the remaining design life of the vessel. These are based on the slope of prediction curves presented in proposed ASTM Standards "Predicting Neutron Radiation Damage To Reactor Vessel Material."

Question (5):

Provide a listing of operator actions which are required for your plant to prevent pressurized thermal shock and to ensure vessel integrity. Include a

description of the circumstances in which these operator actions are required to be taken. Included in this summary should be the specific pressure, temperature and level values for: a) high pressure injection (HPI) termination criteria presently used at your facility, b) HPI throttling criteria and instruction presently used at your facility and c) criteria for throttling feedwater presently used at your facility. For each required operator action, give the information available to the operator and the time available for his decision and the required action. State how each required operator action is incorporated in plant operating procedures and in training and requalification training programs.

RESPONSE (5):

Previous analyses, as well as more recent work by the Westinghouse Owner's Group have confirmed, using conservative assumptions and methods, that a potential reactor vessel integrity problem is not an immediate concern. However, additional detailed analyses of reactor vessel integrity and investigation of remedial actions are also being performed and will be available at the end of this year.

For the low probability accident events that pose a concern, the following are a summary of criteria for operator action relevent to mitigating these events. Analyses performed for the LOCA requires no operator action. The operator action required for the Large Main Steam Line Break is assumed to occur within 10 minutes and we expect this to be a reasonable assumption.

MAIN STEAM LINE BREAK ACCIDENT

Criteria for High Pressure Safety Injection Termination:

The Emergency Operating Procedure for the loss of Secondary Coolant Accident instructs the operators to terminate the HPI if the following conditions are satisfied:

Conditions for early termination of high pressure (1450 psig) safety injection:

- If one wide range reactor coolant temperature T_H (as confirmed by Core Exit Thermocouples) is less than 350°F and
- pressurizer level is above 20% span and
- reactor coolant is above 700 psig and
- subcooling (margin) above 60°F and
- one or more steam generators narrow range level above 15% span or Auxiliary Feedwater Flow of 570 GPM

Criteria for Throttling Auxiliary Feedwater:

When faulted steam generator is identified, isolate auxiliary feedwater flow to faulted steam generator.

When any intact steam generator narrow range level is in the narrow range, control steam generator level by throttling auxiliary feedwater flow.

Criteria for Throttling High Pressure Safety Injection:

None

LOSS OF COOLANT ACCIDENT

Criteria for High Pressure Safety Injection Safety Injection Termination

The Emergency Plant Operating procedures instruct the operators to terminate the HPI in case of LOCA under the following conditions:

- Pressurizer Level is above 50% span and
- Reactor Coolant Pressure above 2000 psig and increasing and
- Subcooling (margin) above 30°F and
- Auxiliary Feedwater Flow of at least 570 gpm or steam generator level greater than 15% narrow range span in one or more steam generators

Criteria for Throttling Auxiliary Feedwater

Only after steam generator water level is established on the narrow range should Auxiliary Feedwater System Flow be reduced to maintain required level.

Criteria for Throttling High Presure Safety Injection

None

The shutoff head of HPI pumps is 1450 psig, thus limiting the repressurization of the reactor vessel.

These actions are taken primarily to assure that under the most adverse conditions, core cooling is maintained thus assuring vessel integrity. Our NSSS vendor has not identified any operational transients that pose a threat to the reactor vessel integrity. The low probability accident scenarios mentioned above pose a concern and will be addressed. As noted above we are participating in the Westinghouse Owner's Group and will modify our procedures based on their results as appropriate.

The following instruments are available in the Control Room to allow the operator to monitor the course of accident and take appropriate action:

REACTOR COOLANT SYSTEM	
	Wide Range (0-700F) Temperature Recorders
VPA-3	Temperature Hot Leg (T_H) A, B, and C
VPA-3	Temperature Cold Leg (T_C) A, B, and C
	Narrow Range (540 - 610F) Temperature Recorder
Console, Vertical Section	T_{AVG} Narrow Range Temperature Recorder
VPA-4	T_{AVG} (Control) Narrow Range Temperature Indicator (540-610F) Loop A, B, and C
	Wide Range (0-2300F) Temperature
Digital data process system, CRT display	8-Core Exit Thermocouples (2 per quadrant)
Pressure	
VPA-2	Wide Range (0-3000 psig) Pressure Indicator
VPA-2	Narrow Range (0-1000 psig) Pressure Indicator
Subcooling (Margin)	
	Digital Indicator
Filler Section Between VPA-11 and VPA-1	Subcooled Margin Monitor
PRESSURIZER	
Pressure	Narrow Range (1700-2500 psig)
VPA-4	Recorder
VPA-3	Pressurizer Indicator Channel I, II, and III

Level	Level (0-100% span)
VPA-3	Level Indicator Channel I, II, and III
STEAM GENERATOR	
Pressure	
	Pressure Indicator (0-1200 psig)
VPA-3	Steam Generator A Channel I, II, and III
VPA-3	Steam Generator B Channel I, II, and III
VPA-3	Steam Generator C Channel I, II, and III
Level	Level Indicator Normal Range (0-100%)
VPA-3	Steam Generator A Channel I, II, and III
VPA-3	Steam Generator B Channel I, II, and III
VPA-3	Steam Generator C Channel I, II, and III
STEAM HEADER	
Pressure	
	Steam Header Pressure Indicator (0-1200 psig)
VPA-3	Steam Header Channel I, II, and III
VPA-3	Steam Header Channel I, II, and III
VPA-3	Steam Header Channel I, II, and III
AUXILIARY FEEDWATER	
Flow	
	Flow Indicators (0-300 GPM)
Console, Vertical Section - 3	Steam Generators A, B, and C
HIGH PRESURE SAFETY INJECTION SYSTEM (HPSIS)	
Pressure	
	Header Pressure
VPB-3	Pressure Indicator (0-2000 psig)
VPB-3	Pressure Indicator (0-2000 psig)

Flow	
	Header Flow
VPB-4	Flow Indicator (0-1000 GPM)
VPB-4	Flow Indicator (0-1000 GPM)
REFUELING WATER STORAGE TANK (RWST)	
	Level Indicator (0-100%)
VPB-3	Level Indicator
CONTAINMENT SUMP LEVEL	Level Recorder (0-100%)
VPA-1	Level Recorder
CONTAINMENT LEVEL	Level Indicator
(Minimum Level 2.93 ft. for ECCS recirculation phase)	
VPB-4	Red Indicating Light Illuminated when 2.93 ft. level is reached.
CONTAINMENT PRESSURE	
VPB-2	Narrow Range (-3" to +3") water gauge
VPB-2	Wide Range (0-80 psig)
CONTAINMENT RADIATION	Radioactivity Levels
Process Radiation	1-100,000 counts per minute Monitor Panel
STEAM GENERATOR SECONDARY SIDE RADIOACTIVITY	Radiation Detector
Process Radiation Monitor Panel	Liquid Sample
CONDENSER RADIOACTIVITY	Radiation Detector
Process Radiation Panel	Steam Jet Air Ejector Outlet
CONTAINMENT TEMPERATURE	Temperature Recorder (0-300F)
VPB-7	Containment Atmosphere

The required operator actions for the steam break accident and the loss of coolant accident are included in Turkey Point Unit No. 4 Emergency Operating Procedures. The contents of the Emergency Operating Procedures are discussed during lecture sessions in training and requalification training programs. During simulator training each operator is required to demonstrate proficiency during simulated emergency conditions involving loss of coolant and steam break accidents.

STATE OF FLORIDA)
) ss.
COUNTY OF DADE)

John A. DeMastry, being first duly sworn, deposes and says:

That he is Manager, Nuclear Licensing of Florida Power & Light Company, the herein;

That he has executed the foregoing document; that the statements made in this said document are true and correct to the best of his knowledge, information, and belief, and that he is authorized to execute the document on behalf of said

John A. DeMastry

Subscribed and sworn to before me this

21st day of October, 1981

Cheryl L. Fredrick

NOTARY PUBLIC, in and for the County of Dade, State of Florida

My commission expires: Notary Public, State of Florida at Large My Commission Expires October 30, 1983 ~~Bonded thru Maynard Bonding Agency~~