RAS J-42

February 11, 2008

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

DOCKETED
USNRC

April 15, 2008 (10:00am)

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

Before the Atomic Safety and Licensing Board Panel

In the Matter of	)	
	)	
Entergy Nuclear Generation Company and	)	Docket No. 50-293-LR
Entergy Nuclear Operations, Inc.	)	ASLBP No. 06-848-02-LR
	)	
(Pilgrim Nuclear Power Station)	)	

ENTERGY'S ANSWER TO BOARD QUESTIONS

Entergy Nuclear Generation Company and Entergy Nuclear Operations, Inc. ("Entergy") provide the following responses to the questions posed by the Atomic Safety and Licensing Board in its January 31, 2008 Order. A Declaration of Stephen J. Bethay is provided in support of these responses.

1. With regard to the Condensate Storage System ("CSS") -

a. What is the minimum leakage rate that is certain to be detectable by the testing of the condensate storage tank ("CST") water level every four hours and, conversely, what is the maximum leakage rate that would not be detected by that testing? Provide a detailed statement of the basis of and sources for your answer.

Response: The purpose of the 4-hour testing of CST water level is to determine that the water level remains above the required water levels, including the 10'-5" reserved for HPCI and RCIC, and not specifically to determine a leak rate. Under normal operation, the level of the CSTs is dynamic (i.e. the CST levels fluctuate as they provide makeup or receive condensate discharge to maintain appropriate condenser water level). Therefore, under normal operation, there is no specific leakage rate that would be detected by or could be readily correlated with the four-hour test results.

U.S. NUCLEAR REGULATORY COMMISSION
In the Matter of Entergy (Pilgrim Nuclear Power Station)
Docket No. 50-293-LR Official Exhibit No. 12
OFFERED by: Applicant/Licensee Intervenor Entergy Exh. 9
NRC Staff Other
IDENTIFIED on 4-10-08 Witness/Panel
Action Taken: ADMITTED REJECTED WITHDRAWN
Reporter/Clerk Thibault

Temp=SECY-028

DS03

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1. With regard to the Condensate Storage System ("CSS") -

a. What is the minimum leakage rate that is certain to be detectable by the testing of the condensate storage tank ("CST") water level every four hours and, conversely, what is the maximum leakage rate that would not be detected by that testing? Provide a detailed statement of the basis of and sources for your answer.

Response: The purpose of the 4-hour testing of CST water level is to determine that the water level remains above the required water levels, including the 10'-5" reserved for HPCI and RCIC, and not specifically to determine a leak rate. Under normal operation, the level of the CSTs is dynamic (i.e. the CST levels fluctuate as they provide makeup or receive condensate discharge to maintain appropriate condenser water level). Therefore, under normal operation, there is no specific leakage rate that would be detected by or could be readily correlated with the four-hour test results.

Under the Pilgrim procedures, normal CST level is maintained between 30 to 38 feet, and corrective action is required if water level drops below 30 feet. Further, 10'-5" of water is maintained solely to support HPCI and RCIC. Therefore, the level would have to drop at least ~19.5 feet before there would be insufficient water to maintain this reserve. Since the water level in each tank is monitored to the nearest foot every four hours, an ~19.5 foot drop would be clearly detectable. Since each foot of water in a CST represents approximately 7,000 gallons, a drop of 19.5 feet would correspond to a loss of 136,500 gallons.

More specifically, each CST is equipped with a level indicator which under Pilgrim procedures is monitored every 4 hours. Further, each CST is equipped with level switches, which trigger an alarm in the control room if water level decreases below 12.5 feet, and trip the condensate transfer pumps (i.e. terminate use for normal operations) if CST level reaches 11.5 feet. If the water level continues to decrease, the HPCI and RCIC suction path is switched (when CST level reaches 8 and 3.5 feet, respectively) to the torus, which is the assured (safety-related) source of water for the HPCI and RCIC functions. These setpoints for swap-over to the torus ensure that net positive suction head to the HPCI and RCIC pumps is maintained.

As stated above, the level indicator measurements cannot be readily correlated with a leak rate during normal operations because the water level is dynamic. However, if a CST were under static conditions (i.e. no makeup or discharge), a level reduction of one foot (corresponding to the minimum change that would normally be recorded) between four hour tests would correspond to a leak rate of 30 gallons per minute.

b. What is the minimum leakage rate that is certain to be detected by the quarterly testing of the water flow from the reactor core isolation cooling ("RCIC") pump and the high pressure cooling injection ("HPCI") pump, and, conversely, what is the maximum leakage rate that *would not be detected by that testing? Provide a detailed statement of the basis of and sources for your answer.*

Response: The quarterly HPCI and RCIC system surveillance tests are not designed or intended to quantify a potential leakage rate from buried piping. Rather, these quarterly tests demonstrate system operability by verifying pump flow rate and discharge pressure. The test is performed by creating a flow path (through the buried pipe) with suction from and discharge to the CST. While these tests do not allow quantification of leakage rate, they do demonstrate that, even if leakage were occurring, the required flow from the CST to HPCI and RCIC would still be achieved.

c. What is the smallest leakage rate that could reasonably be expected to challenge the ability of the CSS system piping at issue to fail to satisfy its intended function(s) as relevant for license renewal? Provide a detailed statement of the basis of and sources for your answer.

Response: At the outset, no amount or rate of leakage from the CSS buried piping could challenge the ability of the HPCI and RCIC systems to perform their intended functions. While the CSTs are the preferred source of water for HPCI and RCIC (because of water purity), the assured (i.e. safety-related) source of water is the torus. If the CSS were unable to deliver water to the HPCI and RCIC pumps, for any reason, the HPCI and RCIC suction path would be switched to the torus.

While leakage from the CSS piping would not prevent the HPCI and RCIC functions from being performed, it could affect the ability of the CSTs to serve as the preferred source of water for HPCI and RCIC. Make-up to the CSTs is supplied from the demineralized water storage tank (DWST). The demineralized water transfer system (DWTS), which transfers water from the DWST to either CST, is supplied by two pumps each of which is rated at 110 gallons per minute. Since only one of the two pumps is normally in service, a maximum of 110 gallons per minute of makeup could be provided to either CST to compensate for a leak. If leakage from buried CSS piping were to exceed this rate, the volume of water in the CST could not be maintained, which would eventually[1] impact its ability to provide the preferred source of water to the HPCI and RCIC systems.

The smallest leakage rate that would challenge the ability of a CST to serve as the preferred source for HPCI and RCIC within a 4 hour interval is on the order of 500 gallons per minute. With regard to a leakage rate that would be detected by the 4 hour monitoring, one could hypothesize the following: Assume initial tank level is at the procedural minimum of 30 feet. A leak develops such that the level drops to the alarm setpoint (12.5 feet) just before the next 4 hour observation. In this case, a level reduction of 17.5 feet over a 4 hour period would represent a leakage rate of over 500 gpm. Because this leakage rate exceeds the make-up capability of the DWTS, the capability of the CST to act as the preferred source would not be recovered without corrective action. However, such a large leakage rate would likely cause visible effects, such as water leaking into the building, erosion of exterior ground surfaces, or significant amounts of visible water in exterior areas, that would be noticeable well within the 4-hour observation period.

[1] For example, if the leakage rate from a CST were 220 gallons per minute (twice the makeup rate of a DWTS pump), it would take about 20 hours before the CST level would be reduced below the level reserved for HPCI and RCIC. The volume of water that would have to be lost to reduce the water level in the CST from its normal minimum (30 feet) to the level reserved for HPCI and RCIC (10.5 feet) is 136,500 gallons ([30 feet – 10.5 feet] x 7,000 gallons/foot). Assuming a single DWST pump provides makeup at its rated capacity, a leak of 220 gallons per minute would correspond to a net loss rate of 110 gallons per minute (220 gallons per minute leakage rate minus 110 gallons per minute makeup rate). The time it would take for this net loss rate to reduce the volume by 136,500 gallons is: 136,500 gallons ÷ [110 gallons per minute x 60 minutes/hour] ~ 20 hours.

Leakage from the buried piping would not be expected to affect the flow of water through the buried CSS line. The positive pressure in the piping would cause any leakage to flow out of the line, not in, so leakage would not be expected to introduce debris or cause blockage of the piping. Further, the key consideration in system operation is maintaining adequate suction pressure (i.e. net positive suction head) to the pumps. The CST and piping system design, in conjunction with the setpoints for swapping the HPCI and RCIC suction source to the torus, ensure adequate net positive suction head to the pumps. Thus, while some amount of water would be diverted from the piping to the ground which would serve to increase the rate of level decrease in the CST, this would merely accelerate the time at which the suction swap to the torus would be required. HPCI and RCIC functions would be unaffected.

2. With Regard to the Salt Service Water ("SSW") system – Explain how any leak in the SSW buried pipes that might carry radioactive water from the plant to the canal that dumps into the bay could challenge the ability of the SSW system to satisfy its intended function(s)? For example, is there any correlation between any potential leak in those pipes and any potential plugs in them that might prevent them from discharging water from the SSW, thereby impeding the ability to remove heat from the RBCCW? Provide a detailed statement of the basis of and sources for your answer.

Response: The SSW system discharge piping is an open-ended run of unobstructed piping. Leakage is generally not a concern for an open-ended discharge pipe.

The external surface of the carbon steel discharge pipe is protected by either a coal tar wrapping or epoxy coating. The interior of the discharge piping is protected by a ½" thick cured-in-place-pipe (CIPP) lining, consisting of polyester felt material with a resin and catalyst system or an epoxy resin and hardener system, which forms a smooth, hard inner protective surface. These coatings and linings are designed to prevent internal and external corrosion. For leakage to occur, there would have to be a failure of the external coating, a through wall failure of the metal pipe, and a failure of the CIPP liner. The likelihood of these three barriers being breached is remote.

Further, in the unlikely event of leakage from the discharge piping, such leakage would not be expected to have any effect on the SSW system's ability to perform its intended function. Leakage would simply result in some salt water being discharged to the ground rather than to the bay. Further, because there is a positive pressure differential within the discharge piping, in-leakage of dirt or debris that might block the discharge line would not be expected. Indeed, even if dirt were introduced, it would likely be swept away with the discharge flow. Moreover, if dirt or debris were somehow accumulating, any significant diminishment of flow would be detected by the daily monitoring of the heat exchange capability of the SSW system. Thus, only if degradation of the SSW discharge piping were somehow to progress to the point of pipe collapse would the SSW system's ability to satisfy its intended function be challenged. The design and construction of the SSW discharge piping, including external coatings and internal liner, make such a failure mechanism not credible.

Respectfully Submitted,

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Dated: February 11, 2008

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(Pilgrim Nuclear Power Station)	)	

DECLARATION OF STEPHEN J. BETHAY IN SUPPORT OF ENTERGY'S ANSWER TO BOARD QUESTIONS

I, Stephen J. Bethay, do hereby state the following:

I am the Director, Nuclear Safety Assurance, for Pilgrim Nuclear Power Station ("PNPS"). My business address is 600 Rocky Hill Road, Plymouth, MA 02360. A statement of my professional qualifications is attached.

I provide this declaration in support of Entergy's answers to the questions asked by the Atomic Safety and Licensing Board in its January 31, 2008 Order. I have knowledge of the matters stated therein, and declare under penalty of perjury that Entergy's answers are true and correct to the best of my knowledge, information, and belief.

Stephen J. Bethay

Stephen J. Bethay

Executed: February 11, 2008

Stephen J. Bethay

Director, Nuclear Safety Assurance

Pilgrim Nuclear Power Station

Responsibility:

Responsible for management and oversight of support functions of the Pilgrim Station including Licensing, Corrective Action Program, Quality Assurance, Emergency Preparedness, and Security,.

Experience:

2004- Present Director, Nuclear Safety Assurance, Pilgrim Station

2001- 2004 Engineering Director, Entergy Nuclear, Pilgrim Station

Responsible for all aspects of engineering support of Pilgrim Station.

1999-2001 Station Services Superintendent, Entergy Nuclear, Pilgrim Station

Responsible for non-power block facility maintenance, radioactive waste shipping, and facility decontamination.

1997-1999 Licensing Director, Entergy Operations Inc., corporate

Responsible for corporate licensing support of the Riverbend, Grand Gulf,

Waterford 3 and Arkansas Nuclear One facilities.

1994-1997 Manager, Engineering, Southern Nuclear Operating Company

Responsible for corporate engineering support including design and project management for

the E. I. Hatch Nuclear Plant

1988-1994 Licensing Manager, Southern Nuclear Operating Company

Responsible for licensing and regulatory support for the E. I. Hatch Nuclear Plant.

1981-1988 Various engineering and licensing positions associated with the E.I. Hatch Nuclear Plant

1977-1981 Co-operative Education Student, Georgia Power Co., Bowen Steam Electric Plant

Education/Training:

(1981) B.S.-Mechanical Engineering, Auburn University

(1987) Station Nuclear Engineer Certification

(2001) SRO Certification, Pilgrim Nuclear Power Station

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CERTIFICATE OF SERVICE

I hereby certify that copies of "Entergy's Answer to Board Questions," dated February 11, 2008, were served on the persons listed below by deposit in the U.S. Mail, first class, postage prepaid, and where indicated by an asterisk by electronic mail, this 11th day of February, 2008.

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