Entergy

Entergy Nuclear Northeast
Entergy Nuclear Operations, Inc.
James A. Fitzpatrick NPP
P.O. Box 110
Lycoming, NY 13093
Tel 315 349 6024 Fax 315 349 6480

T.A. Sullivan
Site Vice President - JAF

July 09, 2005
JAFP-05-0105

Mr. James Dyer
Director, Office of Nuclear Reactor Regulation
United States Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, Maryland 20852

SUBJECT: **James A. FitzPatrick Nuclear Power Plant**
Docket No. 50-333
Request for Approval of Relief Request No. RR-38, Proposed Alternative to Perform a Temporary Non-Code Repair in accordance with 10 CFR 50.55a(a)(3)(ii)

Dear Sir:

Pursuant to 10 CFR 50.55a(a)(3)(ii), Entergy Nuclear Operations, Inc. (ENO) is submitting a relief request (Attachment 1) to perform a temporary non-code repair to that specified by the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, Article IWC-4000.

The Code of Record for James A. FitzPatrick Nuclear Power Plant (JAF) Repair/Replacement Program is the 1989 Edition, No Addenda of the ASME Section XI Code.

JAF is currently in a forced outage to repair a through wall crack in the Torus shell. During the forced outage, plant walkdowns identified a through wall crack at a weld on the common suction header of the Residual Heat Removal (RHR) Shutdown Cooling (SDC) piping. Under the requirements of ASME Section XI, a code repair/replacement is required if the flaw exceeds the acceptable limits. However, SDC is required for decay heat removal when the plant is shutdown. ENO requests relief to perform a temporary non-code repair since a code repair performed under the current plant conditions would present significant hardship and a challenge to safe plant operation without a compensating increase in level of quality and safety, pursuant to 10 CFR 50.55a(a)(3)(ii).

ENO requests approval of the proposed relief request to perform a temporary non-code repair by July 10, 2005, to support the present repair and startup schedule.

There are no new commitments made in this letter. Should you have any questions or comments concerning this submittal, please contact Mr. Rick Plasse at (315) 349-6793.

A047

Sincerely,

T. A. Sullivan
Site Vice President

Attachment: 1. Relief Request No. RR-38, Proposed Alternative to Perform a Temporary Non-Code Repair in accordance with 10 CFR 50.55a(a)(3)(ii)

cc: Regional Administrator, Region I
U. S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406-1415

Office of the Resident Inspector
U. S. Nuclear Regulatory Commission
P. O. Box 136
Lycoming, NY 13093

Mr. John P. Boska, Project Manager
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Mail Stop: O-8-B1
Washington, DC 20555-0001

Mr. Peter R. Smith, President
NYSERDA
17 Columbia Circle
Albany, NY 12203-6399

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555

Mr. Paul Eddy
New York State Department
of Public Service
3 Empire State Plaza, 10th Floor
Albany, New York 12223

Attachment 1 to JAFP-05-0105
Entergy Nuclear Operations, Inc. – FitzPatrick
Docket No. 50-333

Relief Request No. RR-38, Proposed Alternative to Perform a Temporary Non-Code Repair in accordance with 10 CFR 50.55a(a)(3)(ii)

INTRODUCTION

Pursuant to 10 CFR 50.55a(a)(3)(ii), the James A. FitzPatrick Nuclear Power Plant (JAF) requests approval of a relief request to perform a temporary non-code repair to that specified by the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, Article IWC-4000.

JAF is currently in a forced outage to repair a through wall crack in the Torus shell. During the forced outage, a plant walkdown identified a through wall crack at a weld on the common suction header of the Residual Heat Removal (RHR) shutdown cooling (SDC) piping. Under the requirements of ASME Section XI, a code repair/replacement is required. However, SDC is required for decay heat removal when the plant is shutdown, thus preventing a code repair from being performed without a significant hardship to plant operations and potential impact to safety.

A. ASME CODE COMPONENT AFFECTED

The affected piping section is in the common suction header portion of the Shutdown Cooling piping within the RHR system. It is classified as ASME Class 2, Examination Category C-H, Item C7.30 & C7.40.

B. APPLICABLE CODE EDITION AND ADDENDA

The Code of Record for the JAF Repair/Replacement Program is the 1989 Edition, No Addenda of the ASME Section XI Code.

C. APPLICABLE CODE REQUIREMENT

Article IWC-4000 states that the rules of IWA-4000 apply.

Section XI of the ASME B&PV Code (the Code) specifies code-acceptable repair methods for flaws that exceed code acceptable limits in piping that is in service. A code repair is required to restore the structural integrity of the flawed ASME Code piping, independent of the operational mode of the plant when the flaw is detected.

D. REASON FOR REQUEST

JAF is currently in a forced outage to repair a through wall crack in the Torus shell. A plant walkdown identified a through wall crack at a weld attaching a trunion support to the common suction header of the SDC piping within the RHR system. The plant is currently shutdown in Mode 4, Cold Shutdown, with the RHR System lined up in the SDC mode. Although the suction

Relief Request No. RR-38, Proposed Alternative to Perform a Temporary Non-Code Repair in accordance with 10 CFR 50.55a(a)(3)(ii)

line is cracked and the system has been declared inoperable it is currently performing the required decay heat removal function. A temporary support has been installed using the temporary modification process. This temporary support significantly reduces the static loading on the trunion support with the cracked weld. In order to perform a code repair the SDC suction line would need to be removed from service, thus requiring use of an alternative means to remove decay heat.

The most preferable alternative means of removing decay heat is by use of the Main Steam Line (MSL) Drains and the Main Condenser. Reactor Vessel (RV) level is raised to the main steam lines and water is let down to the main condenser via the MSL Drains where it is cooled by circulating water flowing through the main condenser tubes and returned to the RV via the condensate and feed water systems. This alternative method presents the fewest challenges because it uses the normal feed water connections and has little impact on reactor water chemistry. However, by calculation and demonstration, it has been determined that this method only removes approximately half the current decay heat load. Calculations have determined the current heat load to be approximately 24°F/hr heatup. When the MSL drains & Main Condenser line up is in service, an approximate 12°F/hr heatup rate is indicated.

As required by the plant Technical Specifications (TS) there are alternate means of decay heat removal available if needed to maintain the plant in the safe shutdown condition. However, the use of these alternate methods of decay heat removal presents various levels of operational challenges and hardships, as discussed below:

One alternative is to use low pressure Emergency Core Cooling Systems (ECCS) (Core Spray and Low Pressure Coolant Injection (LPCI)) in conjunction with the Safety Relief Valves (SRVs). This alternative method would effectively remove the decay heat, and is credited in the accident analysis of the Updated Final Safety Analysis Report (UFSAR) for this purpose. In this alternative method the RHR System is lined up in the LPCI mode and suction is taken from the torus. Torus water is injected into the RV and returned to the torus via the Safety Relief Valves (SRVs). This method is undesirable due to the upset in normal reactor water chemistry caused by the introduction of torus water into the RV. This flow path also introduces CRUD into the SRVs which results in damage to the valve seats. As a result industry operating experience demonstrates that the service life of the SRVs would be negatively affected after being used in this manner. Due to the high flow rate associated with LPCI this method also presents a challenge to the plant operators in controlling the heat-up and cooldown rates.

Another alternative is to use the feed and condensate systems in conjunction with the SRVs to provide a flow of cool water through the RV. This method has less impact on reactor water chemistry than the previously described method using torus water, but the potential impact on the SRVs is the same. In addition use of this method would require processing and discharging thousands of gallons of liquid radwaste.

The Decay Heat Removal System used during refueling operations is also an alternative. This method requires flooding up and removing the vessel head. This would require expending approximately 10 - 15 person rem for reactor disassembly and reassembly. Use of this method

would require extended use of the SDC system in its degraded condition until the RV could be flooded up, the head and vessel internals removed, the refuel cavity flooded, and conditions for the start-up of DHR established. These conditions include installation of a non-safety related diesel.

In summary, there are available decay heat removal alternatives that would allow isolation of the RHR SDC suction line for a code repair. The use of those alternatives presents challenges and hardships in of terms impact on equipment and plant operations that are not offset by any gain in safety or quality, whereas the proposed non-code repair allows for expedited restoration of the piping integrity with no operational hardships.

E. PROPOSED ALTERNATIVE AND BASIS FOR USE

The proposed alternative is to perform a non-code repair of the flawed weld area by arresting the crack development, stress relieving the crack, and applying a seal weld to the crack. The non-code weld repair will re-establish the integrity of the piping system. Additionally, a temporary support has been installed using the temporary modification process. This temporary support is located within five feet of the existing trunion support and significantly reduces the static loading on the trunion support. The adequacy of the trunion support, assuming no weld in the repair area, to continue to act as a seismic support has been evaluated and found to be acceptable for all loading conditions.

The temporary non-code repair area will be visually inspected by Operations personnel once a day while RHR SDC is in operation to ensure the leak tightness of the weld and piping is maintained.

A code repair in accordance with the requirements of IWC-4000 will be performed when this section of the SDC line can be removed from service. Pressure test of the code repair will be performed in accordance with IWC-5000 requirements.

Basis for Use

The proposed alternative is to perform a temporary non-code repair by arresting the crack development, stress relieving the crack, and applying a seal weld to the crack. This will reestablish the integrity of the piping system. Engineering analysis of the non-code weld repair, including all loading conditions, will be used to establish the operability of the line in a degraded condition. This will support plant mode changes upon completion of the torus code repair and pressure testing. JAF will then proceed with plant startup, thus allowing the SDC system to be removed from service. A code repair on the through wall crack on this section of the SDC piping can then be performed without imposing any operational hardships.

The weld repair will serve to re-establish the leak tightness of the piping system and will be performed utilizing a welding procedure, specification and materials qualified to meet the design

requirements of the RHR piping.

The temporary support that has been installed will alleviate the static loads from the existing trunion support.

In terms of PRA, start up with the SDC suction line in the proposed condition does not result in any change in core damage frequency. JAF's current revision of the IPE specifically accounts for the possibility of a rupture of the RHR SDC piping. This is one of a class of Interfacing System LOCAs (ISLOCA or "V" sequence) specifically evaluated in the PRA model. V sequences are initiated by a failure of the isolation valves that form the pressure boundary between the reactor vessel and low pressure systems. In the postulated worst case scenario, isolation valve failure results in a large break LOCA outside containment.

This is represented in the PRA model for SDC as a 'gate' for RHR SDC ISLOCA Core Damage Sequences. Failure of any of the underlying sequences that feed this 'gate' goes directly to core damage (i.e., cause core damage). The assumption in the PRA model is that if you open 10MOV-17 and 10MOV-18 (RHR SDC system isolation valves) at pressure, the piping system fails. Therefore, the crack in the piping does not affect Core Damage Frequency (CDF). What drives risk is the potential for opening these isolation valves. Adequate administrative controls are in place to control the operation of these valves. Valve operation is controlled through operating procedures, which include valve lineups that would be performed after major valve manipulations or as deemed necessary by the shift manager.

By performing a temporary non-code weld repair the plant can be placed in a condition where the SDC function is no longer required, thus removing the need to use an alternative method. This also allows for plant conditions where safety systems are in service and there are no time constraints on the repair activity. As described in the plants UFSAR, the RHR SDC System is a manually initiated shutdown decay heat removal system. Having it out of service will have no impact on safety, as the appropriate safety systems for removing decay heat during abnormal operational transients will be operable during the repair activity.

F. DURATION OF PROPOSED ALTERNATIVE

This is a one time relief from the requirements of IWC-4000, to perform a temporary non-code repair seal weld.