Entergy

Entergy Nuclear Operations, Inc.
Pilgrim Station
600 Rocky Hill Road
Plymouth, MA 02360

William J. Riggs
Director, Nuclear Assessment

February 3, 2003

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: Entergy Nuclear Operations, Inc.
Pilgrim Nuclear Power Station
Docket No. 50-293
License No. DPR-35

Pilgrim Relief Request (PRR)-28, Revision 1,
Relief from ASME Code, Section XI, Examinations of Reactor Pressure Vessel Circumferential Shell Welds, Pursuant to Generic Letter 98-05.

REFERENCE:

1. Entergy Letter No. 2.02.059, Request for Relief Request (PRR)-28, Relief from ASME Code, Section XI, Examination of Reactor Pressure Vessel Circumferential Welds, Pursuant to Generic Letter 98-05, dated July 1, 2002.

2. NRC Letter, "Request for Additional Information Regarding Pilgrim Relief Request (PRR)-28 (TAC NO. MB6074), dated January 22, 2003

LETTER NUMBER: 2.03.009

Dear Sir or Madam:

Entergy requested NRC review and approval of Pilgrim Relief Request No. 28 (Reference 1) in support of refueling outage (RFO)-14. PRR-28 sought relief from ASME Code, Section XI, Examination of Reactor Pressure Vessel Circumferential Welds, pursuant to Generic Letter 98-05. Subsequent discussions with the NRC staff identified additional information needed in the submittal to support the NRC's review. This letter provides the information discussed with the NRC staff in response to the staff's request for additional information (Reference 2) and replaces the original submittal. The additional information supports the conclusions in Reference 1 that the alternatives proposed provide an acceptable level of quality and safety as required by 10 CFR 50.55a(a)(3)(i).

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If you have any questions regarding the information contained in this letter, please contact Mr. Bryan Ford (508) 830-8403.

Sincerely,

W.J. Biggs

Attachments:
1. Description of Pilgrim Relief Request No. 28 (10 pages)
2. Pilgrim Relief Request No. 28 (2 pages)
3. Response to NRC Request for Additional Information (1 Page)

cc: Mr. Travis Tate, Project Manager
Office of Nuclear Reactor Regulation
Mail Stop: 0-8B-1
U. S. Nuclear Regulatory Commission
1 White Flint North
11555 Rockville Pike
Rockville, MD 20852

U.S. NRC, Region 1
475 Allendale Road
King of Prussia, PA 19406

Senior Resident Inspector
Pilgrim Nuclear Power Station

Attachment 1

DESCRIPTION OF
PILGRIM RELIEF REQUEST-28

A. RELIEF REQUEST

Pursuant to 10CFR50.55a(a)(3)(i) and consistent with information contained in NRC Generic Letter (GL) 98-05 (Reference 1), Pilgrim is requesting permanent relief (for the remaining portion of the initial license period that expires in 2012) from ASME Section XI requirements to examine Category B-A reactor pressure vessel (RPV) circumferential shell welds.

B. EXAMINATION REQUIREMENTS

10CFR50.55a(g) requires examination of Reactor Pressure Vessel (RPV) shell welds specified in item B1.10 of Examination Category B-A, "Pressure Retaining Welds in Reactor Vessel", in Table IWB-2500-1 of the 1989 Edition of Section XI, Division 1, of the ASME Boiler and Pressure Vessel Code for each ISI interval.

C. ALTERNATIVE TO THE EXAMINATION REQUIREMENTS

As an alternative to the examination requirements of circumferential shell welds, Pilgrim proposes the following:

1. The examinations of essentially 100% of accessible RPV axial welds to the extent possible and essentially 1 to 3% of the incidental intersecting circumferential welds during RFO-14 and/or 15 (Third ISI interval).

2. Compliance with GL 98-05 provisions included in the BWRVIP-05 safety evaluation.

D. BASIS FOR THE ALTERNATIVE PLAN

1. Pilgrim complied with 10CFR50.55a(g)(6)(ii)(A)(2) required augmented examination of RPV circumferential and axial shell welds during RFO-10, in the Second ISI interval that ended June 30, 1995. Pilgrim provided the results of these examinations to the NRC (Reference 2). The NRC issued a safety evaluation of the Pilgrim augmented examination results (Reference 3). The examinations detected no flaws in the vessel. The RFO-10 scope of examinations included essentially all-accessible welds.

2. Pilgrim will comply with the 1989 ASME Code, Section XI requirements to examine essentially 100% of all accessible axial welds and essentially 1 to 3% of incidental intersecting circumferential welds in accordance with 10CFR50.55a(g). These examinations are planned for RFO-14 and/or 15. Pilgrim will exclude the RPV circumferential welds from the RFO-14 and/or 15 examination scope and permanently defer them based on the approved relief, as provided by GL 98-05.

3. Pilgrim complies with GL 98-05 provisions for seeking relief from the requirements of examination of BWR RPV circumferential shell welds as recommended in the BWRVIP-05 Report (Reference 4). GL 98-05 states that licensees may request relief from the examination of the RPV circumferential shell welds by demonstrating the following:

   a. At the expiration of the license, the circumferential welds will continue to satisfy the limiting conditional failure probability for circumferential welds in the staff's July 30, 1998, safety evaluation related to BWRVIP-05 (Reference 3).

   b. Licensees have implemented operator training and established procedures that limit the frequency of cold over-pressure events to the amount specified in the staff's July 30, 1998, safety evaluation related to BWRVIP-05.

   Pilgrim has performed an assessment of the above GL 98-05 provisions, as discussed below.

E. PILGRIM ASSESSMENT OF GL 98-05 PROVISIONS

1. The following demonstrates that at the expiration of Pilgrim Operating License in 2012, Pilgrim RPV circumferential shell welds will continue to satisfy limiting conditional failure probability for the circumferential welds stated in the Staff's July 30, 1998 evaluation (Reference 3).

   a. Neutron Fluence/Embrittlement

      BWRVIP-05 stated, *"Embrittlement issues are addressed in 10CFR50 Appendix G through requirements associated with upper shelf energy (USE) and the reference temperature of nil-ductility transition (RT_{NDT}). In order to account for the effects of embrittlement, adjusted reference temperatures (ARTs), defined as the initial RT_{NDT} plus the irradiation shift for fluence, are determined. It is possible that ARTs may result in pressure-temperature testing criteria that are difficult to meet due to increased temperature requirements. However, due to low BWR fluence, an unacceptable ART will not be reached, even when extended life is planned."* Also, the report states *that "In addition to increasing RT_{NDT} the USE of low alloy steel materials decreases with neutron exposure. However, for the relatively low fluence BWR, maintaining a USE above 50 ft-lbs is not a concern. Also, Code margins required by Appendix G are satisfied at USE values as low as 35 ft-lbs and thus is not a safety concern. Based on the above, it can be seen that although irradiation embrittlement of materials can be a significant concern, its effect is minimal for the relatively low fluence environment of a BWR."*

      The Pilgrim minimum USE at end-of-license is 59 ft-lbs, which exceeds the upper bound allowable limit of 50 ft-lbs specified in 10 CFR 50, Appendix G, section IV.A.1.a. Thus, the 59 ft-lbs USE provides margin of safety against fracture; therefore, neutron fluence / embrittlement is not a safety concern for Pilgrim vessel.

b. Probabilistic Fracture Mechanics (PFM) Analysis

Although BWRVIP-05 provides a technical basis for this relief, an independent NRC risk informed assessment of the analysis contained in the BWRVIP-05 report was conducted (Reference 6). The independent NRC assessment used the FAVOR code to perform probabilistic fracture mechanics (PFM) analysis to estimate RPV failure probabilities. Three key assumptions in the PFM analysis are: the neutron fluence was estimated to be end-of-license mean fluence, the chemistry values are mean values based on vessel types, and the potential for beyond design basis events were considered.

The following is a statement contained in the "Executive Summary" of the "NRC Staff Final Safety Evaluation of BWRVIP-5 Report (Reference 5*). "It should be noted that the failure frequency for axial welds cited above are relatively high, but that there are known conservatisms in these estimates. For example, these analyses were based on the assumption that the flaws in axial weld with the limiting material properties and chemistry are all located at the inside surface of the BWR RPV and at the location of peak end-of-license (EOL) azimuth fluence. Since flaws are distributed throughout the weld and EOL neutron fluence will not occur for many years, the staff has concluded that the present RPV failure frequency is substantially below that reported by the BWRVIP, and independently calculated by the staff, and is not a near-term safety concern."*

The following information is provided to show the conservatism of the NRC analysis with respect to the Pilgrim plant. Changes in RT_{NDT} may be used as one of the means for monitoring radiation embrittlement of reactor vessel materials. For plants with RPVs fabricated by Combustion Engineering (CE), the mean end-of-license neutron fluence for circumferential welds used in the NRC staff and BWRVIP Limiting Plant-Specific Analysis (32 EFPY), Table 2.6-4 of the Safety Evaluation for BWRVIP-05, was 2.0E + 18 n/cm^2. However the highest fluence anticipated for Pilgrim belt-line circumferential wells at the end-of-license (32 EFPY) is 8.03E + 17 n/cm ^{2}. The projected fluence for the Pilgrim plant for 32 EFPY is less than that used in the NRC analysis. Therefore, there is significant conservatism, with regard to the effect of fluence on embrittlement, in the already low circumferential weld failure probabilities as related to the Pilgrim Plant.

The Table below shows a comparison between the NRC Final Evaluation of the BWRVIP-05 Limiting Plant Specific Analysis data and Pilgrim specific data for weld chemistry factor (CF), adjustment of the reference temperature (ΔRT_{NDT}), and mean RT_{NDT}.

Pilgrim RPV Shell Weld Information
Bounding Circumferential Weld (1-344)
Wire Heat/Lot (21935/3869)

Parameter Description	Pilgrim NPS Shell Bounding Beltline 32 EFPY Bounding Comparative Parameters (Bounding Circ. Weld)	USNRC Limiting 32 EFPY Bounding CE Vessel Parameters SER Table 2.6-4
Neutron fluence at the end of the requested relief period (upper bound value @ ¼ t))	8.03×10^{17} n/cm^2	2.0×10^{18} n/cm^2
Initial (unirradiated) reference temperature (RT_{NDT}), °F	-50	0
Weld Chemistry factor (CF), °F	172.2	172.2
Weld Copper content %	0.183	0.183
Weld Nickel content %	0.704	0.704
Increase in reference temperature (ΔRT_{NDT}), °F	64.5	98.1
Mean adjusted reference temperature (ART), °F $RT_{NDT(u)} + \Delta RT_{NDT}$	14.5	98.1

As shown above, the impact of irradiation results in lower plant-specific mean RT_{NDT} for the Pilgrim circumferential weld material, as compared to that for any of the Staff's plant-specific analyses that were performed for the CE fabricated RPV's with the highest adjusted reference temperatures. Therefore, based on plant specific data, there is a lower conditional probability of failure for circumferential welds at Pilgrim than that stated in the NRC's Final Safety Evaluation of the BWRVIP-05.

2. The following demonstrates that Pilgrim has implemented operator training and established procedures that limit the frequency of cold over-pressure events to the amount specified in the staff's July 30, 1998, safety evaluation.

   At an industry meeting on August 8, 1997, the NRC indicated that the potential for, and consequences of, non-design basis events not addressed in the BWRVIP-05 report should be considered. Later, in a Request for Additional Information (RAI) to the BWRVIP, the NRC requested that the BWRVIP evaluate the potential for non-design basis cold over-pressure transients (Reference 6) and responded to in BWRVIP letter to NRC dated December 18, 1997 (Reference 7). The NRC also considered beyond design basis events, such as low temperature over-pressure (LTOP) events in their PFM analysis. In the BWRVIP responses to the RAI the total probability of an occurrence of cold overpressure for other than BWR-4s was reported as 9E-4. It was concluded that it is highly unlikely that a BWR would experience a cold over-pressure transient. In fact, for a BWR to experience such an event would generally require several operator errors. The NRC described several types of events that could be precursors to BWR RPV cold over-pressure transients. These were identified as precursors because no cold over-pressure event has occurred at a U.S. BWR. Also, the NRC identified one actual cold over-pressure event that occurred during shutdown at a non-U.S. BWR. This event apparently included several operational errors that resulted in a maximum RPV pressure of 1150 psi with a temperature range of 79° to 88° F.

The following addresses the high-pressure injection sources, administrative controls, and operator training regarding a cold overpressure event for the Pilgrim plant.

a. <u>Review of Potential High Pressure Injection Sources</u>

1. High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) Systems

   The HPCI and RCIC systems use steam driven turbines to pump cold water into the Pilgrim vessel. During reactor cold shutdown condition, there is no steam available to operate these systems, making a cold over pressurization event impossible as the result of operation of these systems.

2. Feedwater / Condensate Systems

   The feedwater / condensate systems are a potential high-pressure injection water sources into the reactor vessel. The condensate pumps provide water source to the reactor feed pumps. The feed pumps provide water to the vessel. The reactor feed pump discharge pressure is 1200 psig and the shutoff head pressure is 1300 psig. The condensate pump discharge pressure is 450 psig and the shutoff head is 600 psig. A system design feature of the reactor feed pumps is an automatic trip of all feed pumps on high vessel water level.

   The startup and cool down procedure requires detailed monitoring of reactor vessel temperatures and pressures until the coolant temperature is greater than 212°F and vessel metal temperatures are acceptable for critical core operations in accordance with the P-T curves. The condensate and feed water pumps are used to control vessel level during startup. The reactor head vents are not closed until the coolant temperature is greater than 212°F and vessel metal temperatures are acceptable for critical core operations. This administrative action for head vent closure serves as a mechanism to reduce the likelihood of pressurization above 310 psig. When shutting down, Pilgrim procedures require securing RFPs in sequence depending on the reactor power levels. Monitoring of reactor temperature, pressure, and cool down rates, are prescribed in procedures and Technical Specifications. During refueling outages the feedwater lines are isolated by closing block valves inside the drywell. At low power (approximately <10%), the lines are secured by removing both regulating valves from service and manually controlling feed water with a startup-regulating valve.

   Reactor over pressurization by the feedwater/condensate systems is very unlikely since strict controls on temperature and pressure are imposed below 450° F and the capacity of the systems to inject water is limited by using the feedwater startup-regulating valve. Any unexpected change in reactor water level would allow for operator action.

   Therefore, these systems do not present a significant potential for over pressurization.

3. Standby Liquid Control System

   The Standby Liquid Control System (SLC) is a potential source of high-pressure water into the RPV during cold shutdown conditions. The system does not have an automatic start function. A key lock switch in the control room controls this system, with the key normally removed. As a result, operation of SLC is a deliberate act strictly controlled by plant procedures and training. Even if the system was activated the maximum SLC flow is 40 gallons per minute, a rate that would allow time to control RPV pressure. Therefore, this system does not present a significant potential for over pressurization.

4. The Low-Pressure Coolant Injection, Core Spray, and Residual Heat Removal Systems

   The Low-Pressure Coolant Injection, Core Spray, and Residual Heat Removal systems' inadvertent operation do not present a significant potential for over pressurization. The pressure-temperature curves for the Pilgrim RPV permit cooling water pressures up to 310 psig over the temperature range of 70 to 100 degrees F and rapidly increase to 660 psig at 100° F. Shutoff head pressure for a Core Spray pump is nominally 350 psig, shutoff head for an RHR pump is nominally 332 psig. Since these shut-off head pressures could pressurize the reactor vessel to greater than 310 psig, a review of LPCI, RHR and Core Spray operating modes is provided.

   The reactor vessel is not likely to pressurize under LOCA conditions. The LPCI and Core Spray would be actuated with the reactor in a depressurized but metal hot condition during a LOCA from full power, which rapidly depressurizes the reactor vessel to a depressurized but metal hot condition. The metal temperature lags pressure substantially and would be greater than 100°F.

   In an emergency, following a loss of shutdown cooling, an alternate shutdown-cooling mode is permitted. This mode of shutdown cooling uses a CS pump to circulate water from the condensate storage tanks to the torus, using a flow path through the reactor vessel and SRV discharge lines. The reactor head vents are closed and the SRV control switches are placed in the "OPEN" position. With the SRV control solenoids energized, when pressure at the SRV main discharge reaches approximately 50 psig, the SRV will open allowing coolant to exit the reactor vessel and flow to the torus via the SRV discharge lines. In this situation, the open SRVs prevent reactor pressure from exceeding 310 psig. This mode of shutdown cooling is used only in emergency situations and is permitted only when the P-T limit for subcritical heatup and cooldown can be maintained.

The normal testing of LPCI and Core Spray at power and shutdown uses flow path of suction from the torus with return to the torus. The testing flow path will not enter the reactor vessel because there are two valves in-series that are interlocked to prevent simultaneous opening with reactor pressure above 400 psig. In a shutdown condition with either system under test lineup, an inadvertent injection would be detected by operations and the injection would be terminated before exceeding 310 psig based on observation and alarm of reactor vessel level. If an RHR pump or core spray pump were used for reactor vessel fill with the vessel head installed the reactor would have been in the cold and vented condition. The reactor head vents would prevent reactor vessel pressurization to above 310 psig until operator action secured the injection path. Procedures are in place to perform a core spray injection in the shutdown condition to test the full flow capability of the Core Spray injection check valves. Under this procedure, injection flow duration is procedurally limited to that which will indicate a reactor vessel level change and the operators are instructed to minimize vessel level rise. This administrative action would prevent reactor pressure from exceeding 310 psig. A Core Spray pump may be used for reactor vessel and cavity fill during the outage. Under these conditions the vessel head vent will be open when the vessel is filled to the flange. Subsequent operations to fill the vessel cavity will be open to atmosphere and prevent over pressurization.

The shutdown-cooling mode of the RHR system has design prevention from over pressurization since the RHR suction valves from the reactor will close at pressure greater than 70 psig. Closure of the suction valves will trip the running RHR pump preventing vessel pressures from approaching 310 psig.

The reactor head vents are not closed until the coolant temperature is greater than 212°F and vessel metal temperatures are acceptable for critical core operations. This administrative action for head vent closure serves as a mechanism to reduce the likelihood of pressurization above 310 psig.

Therefore, these systems do not present a significant potential for over pressurization.

5. Control Rod Drive (CRD) and Reactor Water Clean-up (RWCU) Systems

   The CRD and RWCU systems are used to control RPV water level and pressure during cold shutdown conditions using a feed and bleed process. Reactor pressure is controlled by opening the reactor head vents when the reactor coolant temperature is less than 220 degrees F. The low flow rate of these pumps (CRD 50 gpm, RWCU 222 gpm) allows sufficient time for operator action to occur and react to unanticipated level changes. Therefore, these systems do not present a significant potential for over pressurization.

6. Class 1 Pressure Test

The procedure used at Pilgrim to perform hydrostatic testing incorporates controls, limitations, and precautions that are reviewed by all personnel involved with maintaining RPV pressure/temperature controls. Rigorous abort criteria provides for immediate actions when limits are approached. An assigned Senior Reactor Operator Control Room Supervisor controls the total procedural evolution and a Test Director assures the procedure is coordinated from start to finish. The procedure is only initiated after a detailed pre-job briefing of all affected people at which time all procedural precautions, limitations, and abort criteria are presented. RPV temperature and pressure are monitored throughout the test to ensure compliance with required pressure-temperature limits. Pressurization rates are controlled through out the test and direction is provided for control using the RWCU or the CRD pumps. Therefore, this system test does not present a significant potential for over pressurization.

In conclusion, the review of potential high pressure injection sources confirms that a cold overpressure event at Pilgrim is extremely unlikely.

b. Review of Operator Training and Work Control Process

1. Reactor Operator Training

Licensed Operator Training provides another method to control reactor water level, temperature, and pressure, in addition to the design and procedural barriers discussed above. Simulator training for start-up and shut down scenarios provides opportunities to perform these operator actions to maintain reactor pressure-temperature limits.

Procedural controls for reactor temperature, water level, and pressure are an integral part of Operator training. Specifically, operators are trained in methods of controlling RPV water level within specified limits, as well as responding to abnormal RPV water level conditions outside the established limits. Additionally, control room operators receive training on brittle fracture limits and compliance with the Technical Specification pressure-temperature limits curves. Plant-specific procedures have been developed to provide guidance to the operators regarding compliance with the Technical Specification requirements on pressure-temperature limits.

2. Work Control Process

During plant outages, work control procedures require that the outage schedule and changes to the schedule receive a risk assessment review commensurate with their safety significance. Senior Reactor Operators provide input to the outage schedule to avoid conditions that could adversely impact reactor water level, pressure, or temperature. Schedules are issued listing the work activities to be performed.

During refueling outages, work is coordinated through the Outage Control Center. In the Control Room, the Shift Manager is required, by procedure, to maintain cognizance of any activity that could potentially affect reactor water level or decay heat removal. The Control Room Operator is required to provide positive control of reactor water level and pressure within the specified bands, and promptly report when operating outside the specified band, including restoration actions being taken. Cognizant individuals involved in the work activity attend pre-job briefings. Expected plant responses and contingency actions to address unexpected conditions, or responses that may be encountered, are included in the briefing discussion.

Based upon the above, the probability of a low temperature RPV over-pressure event at Pilgrim is considered to be less than or equal to that used in the USNRC safety evaluation.

3. <u>Conclusion</u>

Deferral of the RPV circumferential shell weld examinations to the end of the current operating license does not impact quality and safety as discussed above. Entergy believes a permanent deferral in performing the augmented inspections of the RPV circumferential shell welds for the remaining operating license of the plant provides an acceptable level of quality and safety.

F. ALTERNATIVE EXAMINATIONS

The circumferential weld examinations would be permanently deferred for the remainder of the current operating license of the plant. The alternative plan would require examination of RPV axial welds based on accessibility.

This relief request and the alternative method is consistent with the GL 98-05 provisions as described above, and provides an acceptable level of quality and safety as required by 10 CFR 50.55a(a)(3)(i).

G. REFERENCES

1. NRC Generic Letter 98-05, "Boiling Water Reactor Licensee Use of the BWRVIP-05 Report to Request Relief From Augmented Examination Requirements on Reactor Pressure Vessel Circumferential Shell Welds," dated November 10, 1998.

2. Pilgrim Letter 95-099, "Results of Augmented Examination of the RPV Shell Welds and Relief Request Pursuant to 10CFR50.55a(g)(6)(ii)(A)," dated September 20, 1995.

3. NRC Safety Evaluation, "Pilgrim Nuclear Power Station – Request for Authorization of Alternative Reactor Pressure Vessel Examinations," (TAC. No. M93724), dated March 26, 1996.

4. EPRI TR-105697, "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)," September 1995.

5. NRC Letter from Gus C. Lainas, Acting Director, Division of Engineering, Office of Nuclear Reactor Regulation, to Carl Terry, BWRVIP Chairman, Niagara Mohawk Company, "Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-05," Report, dated July 28, 1998.

6. NRC Letter from Brian W. Sheron, Director, Division of Engineering, Office of Nuclear Reactor Regulation, to Carl Terry, BWRVIP Chairman, Niagara Mohawk Company, "Transmittal of NRC Staff's Independent Assessment of the Boiling Water Reactor Vessel and Internals Project BWRVIP-05 Report and Proprietary Request for Additional Information," dated August 14, 1997.

7. BWRVIP Letter, Carl Terry, BWRVIP Chairman, Niagara Mohawk Company, to the NRC, C.E. Carpenter, "BWRVIP Response to NRC Request for Additional Information on BWRVIP-05", dated December 18, 1997.

Attachment 2

PILGRIM RELIEF REQUEST NO. 28

RELIEF REQUEST NUMBER: PRR-28

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THIRD INTERVAL RELIEF REQUEST

SYSTEM/COMPONENT(S) FOR WHICH RELEIF IS REQUESTED

ASME Code Class: 1

System: Reactor Pressure Vessel (RPV)

Components: RPV Circumferential Shell Welds

ASME SECTION XI REQUIREMENTS:

ASME Section XI, 1989 Edition, Subsection IWB, Table IWB 2500-1, Examination Category B-A, Item No. B1.11, and the augmented examination requirement of 10CFR50.55a(g)(6)(ii)(A)(2) requires volumetric examination of essentially 100% of RPV circumferential weld and base material regions in the reactor pressure vessel each inspection interval.

BASIS FOR RELIEF REQUEST:

Pursuant to 10CFR55.55(a)(3)(i), and consistent with information contained in NRC Generic Letter 98-05, Pilgrim is requesting an alternative from ASME Section XI requirements to examine essentially 100% of accessible Category B-A circumferential welds and is proposing permanent relief (for the remaining portion of the initial license period) from these examinations.

Pilgrim completed the first augmented RPV shell weld examination in 1995 during RFO 10. Refer to BECO Letter 95-099. This examination included both horizontal and vertical RPV shell weld to the extent possible and detected no flaws within the criteria of ASME XI IWB-3500.

Consistent with the NRC Generic Letter 98-05 the following is provided.

1. At expiration of license, the circumferential welds will continue to satisfy the limiting conditional failure probability for circumferential welds in the staffs July 30, 1998 safety exclusion. The NRC evaluation of BWRVIP-05 utilized the FAVOR code to perform a probabilistic fracture mechanics analysis to estimate RPV failure probabilities. Although BWRVIP-05 provides the technical basis supporting the relief request, the following information is provided that shows Pilgrim vessel is enveloped by the NRC analysis.

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Pilgrim RPV Shell Weld Information
Bounding Circumferential Weld (1-344)
Wire Heat/Lot (21935/3869)

Parameter Description	Pilgrim RPV Shell Bounding Beltline 32 EFPY Bounding Comparative Parameters (Bounding Circumferential. Weld)	USNRC Limiting32 EFPY Bounding CE Vessel Parameters SER Table 2.6-4
Neutron fluence at the end of the requested relief period (upper bound value @ ¼ t))	**8.03×10^{17} n/cm^2**	**2.0×10^{18} n/cm^2**
Initial (unirradiated) reference temperature (RT_{NDT}), °F	**-50**	**0**
Weld Chemistry factor (CF), °F	**172.2**	**172.2**
Weld Copper content %	**0.183**	**0.183**
Weld Nickel content %	**0.704**	**0.704**
Increase in reference temperature (ΔRT_{NDT}), °F	**64.5**	**98.1**
Mean adjusted reference temperature (ART), °F $RT_{NDT(u)} + \Delta RT_{NDT}$	**14.5**	**98.1**

2. Pilgrim has implemented Operator training and established procedures that limit the frequency of cold over-pressure events to the amount specified in the staffs July 30, 1998 safety evaluation.

ALTERNATIVE:

All longitudinal (axial) RPV shell welds, Code Item B1.11, will be examined to the extent possible. If less than 90% coverage is achieved we shall request additional relief. In addition, essentially 1 to 3% of the incidental circumferential welds will be examined during RFO-14 and/or RFO-15 (Third ISI interval).

APPLICABLE TIME PERIOD

Relief is requested for the remaining portion of the initial Operating License period that expires in 2012. This includes 3rd and 4th ISI intervals.

ATTACHMENT 3

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION

NRC QUESTION No. 1

In addressing the BWRVIP-05 report, Generic Letter (GL) 98-05 states "... perform inservice inspections (ISI) on essentially 100 percent of the RPV axial shell welds, and essentially zero percent of the circumferential RPV shell welds, except for the intersections of axial and circumferential welds," in regards to the scope of the inspections. PRR-28 does not provide information on the percentage of intersecting circumferential welds that will be inspected as a result of axial weld inspections. What percentage of circumferential welds will be inspected under Pilgrim's proposal?

RESPONSE

Pilgrim plans to inspect 1 to 3% of the incidental circumferential welds. Sections C.1 and D.2 of Attachment 1 are revised accordingly.

NRC QUESTION No. 2

In Section E.2 (a)(4) of Attachment 1, the licensee indicated that the LPCI, CS, and RHR systems do not operate at pressures above 400 psig. Considering that the Pilgrim P/T curves limit pressure to 310 psig over the temperature range of 70 to 100 degrees F, provide additional details on the controls to prevent a over-pressure event by the LPCI, CS, and RHR systems below 100 degrees F?

RESPONSE

Section E.2 (a)(4) of Attachment 1 is revised and provides additional details on the controls to prevent over-pressure events by the LPCI, CS, and RHR systems below 100° F.

NRC QUESTIONS No. 3 and 4

Provide additional details on controls to prevent inadvertent injection by reactor feed pump or condensate pumps below 450 degrees F.

What are the discharge pressures of the feed water pumps and condensate pumps?

RESPONSE

Section E.2 (a)(2) is modified and provides response the above questions.

NRC QUESTION No. 5

Are there any automatic starts to the SLC system?

RESPONSE

Section E.2 (a)(3) is revised to state that SLC does not have an automatic start function.