Entergy

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January 16, 2004

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

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

Request for NRC Approval of Engineering Evaluation of Elevated Safety Relief Valves' Discharge Pipe Temperatures

LETTER NUMBER: 2.04.004

Dear Sir or Madam:

In accordance with 10 CFR 50.90 and Pilgrim Station Technical Specification 3.6.D.4, Entergy Nuclear Operations, Inc. (Entergy) requests NRC approval of the engineering evaluation of elevated discharge pipe temperature of main steam safety relief valves (SRV) RV-203-3A and RV-203-3D (SRV-3A and SRV-3D).

NRC approval is requested by April 8, 2004 to avoid a shutdown of Pilgrim Station.

Enclosure 1 provides the evaluation of this request. Enclosure 2 identifies the commitments contained in this letter.

Please feel free to contact Bryan Ford, (508) 830-8403, if you have any questions or require additional information.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 16 day of January 2004.

Sincerely,

Michael A. Balduzzi

Michael A. Balduzzi

Enclosure 1: Evaluation of Request (15 pages)
Enclosure 2: Commitments (1 page)

A001

204004

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ENCLOSURE 1

Evaluation of Request

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## Evaluation of Request

### 1. DESCRIPTION

The temperature indicators for two Safety Relief Valves (SRVs) are indicating increased temperatures. The instruments are associated with the discharge pipes (tailpipes) for SRV-3A and SRV-3D.

Pilgrim Nuclear Power Station Technical Specifications Section 3.6.D.3 states:

> *If the temperature of any safety relief valve discharge pipe exceeds 212°F during normal reactor power operation for a period greater than 24 hours, an engineering evaluation shall be performed justifying continued operation for the corresponding temperature increases.*

Pilgrim Nuclear Power Station Technical Specifications Section 3.6.D.4 states:

> *Power Operation shall not continue beyond 90 days from the initial discovery of discharge pipe temperatures in excess of 212°F for more than 24 hours without prior NRC approval of the engineering evaluation delineated in 3.6.D.3.*

The discharge pipe temperature for SRV-3A exceeded 212°F on January 9, 2004 and by 0035 hours on January 10th, the discharge pipe temperature had exceeded 212°F for greater than 24 hours. The discharge pipe temperature for SRV-3D exceeded 212°F on January 11, 2004 and by 1735 hours on January 12th, the discharge pipe temperature had exceeded 212°F for greater than 24 hours.

### 2. REQUESTED APPROVAL

Pilgrim requests NRC approval of this evaluation that it is safe to continue to operate with the leakage past SRV-3A and SRV-3D as indicated by the discharge pipe temperatures greater than 212°F. This approval is requested in accordance with 10 CFR 50.90 and Technical Specification 3.6.D.4, and is requested by April 8, 2004 to avoid a shutdown of Pilgrim Station.

Pilgrim commits to enforce the following limits on continued operation with higher discharge pipe temperatures for SRV-3A and SRV-3D.

> If the discharge pipe temperature for either SRV exceeds 235°F for 24 hours an orderly shutdown of the reactor shall commence and the reactor pressure shall be less than 104 psig within 24 hours. In addition, if the discharge pipe temperature for either SRV exceeds 250°F an orderly shutdown of the

reactor shall commence and the reactor pressure shall be less than 104 psig within 24 hours.

Technical Specification surveillance 4.6.D.3 requires that SRV discharge pipe temperature be logged daily. This surveillance shall be performed at an increased frequency of once per hour while the discharge pipe temperatures are greater than 212°F for SRV-3A and SRV-3D.

## 3. TECHNICAL ANALYSIS

### 1. Problem Statement

Safety Relief Valves SRV-3A and 3D are believed to be leaking. For SRV-3D this condition was detected by discharge pipe temperature monitoring instrumentation on 12/22/03. The SRV-3D discharge pipe temperature trended up to 135 °F on 12/22/03, and to 150 °F on 12/29/03. It is currently at approximately 212°F and is expected (based on past experience) to stabilize between 215°F to 220°F. The current temperature profile supports a condition indicative of pilot stage leakage.

For Safety Relief Valve SRV-3A this condition was detected by discharge pipe temperature monitoring instrumentation on 1/8/04. The SRV-3A discharge pipe temperature trended from 125°F to 215°F in a period of 5 hours. It is currently stabilized at approximately 212°F and is expected (based on past experience) to stabilize between 215°F to 220°F. The current temperature profile supports a condition indicative of pilot stage leakage; however, in this case the leakage may be the result of a vibration or thermal induced anomaly, which caused a sudden increase in temperature. This same condition occurred with the SRV-3A in September 1993 when a sudden increase in discharge pipe temperature occurred but returned to normal after a short period of time. Subsequent, inspections noted degraded insulation on the valve body with gaps present in some areas. Testing at the SRV test laboratory was able to reproduce this unusual leakage condition by loosening pieces of insulation while the valve was on the test stand. In both cases, pilot leakage tends to stabilize and/or slowly increase over a long period of time (12-18 months). During this time the valves are expected to remain in the operable temperature band of less than 235°F.

Technical Specification 3.6.D.3 requires an engineering evaluation to support continued operation if the temperature of any safety relief valve discharge pipe exceeds 212°F for a period greater than 24 hours during normal reactor power operation (Reference 1). The Technical Specification Bases states that minimal leakage exists when the discharge pipe temperature is 215°F, and therefore, a conservative temperature limit of 212°F was chosen.

2. Assumptions

Following an evaluation of the SRV-3D and SRV-3A temperature profiles it has been determined that the leakage is most likely due to pilot stage (pilot seat area) leakage as explained in the evaluation section.

3. Engineering Evaluation

The safety relief valves are part of the reactor coolant pressure boundary and operate by power actuation (i.e., automatic depressurization system) or self-actuation by process high pressure. The safety relief valves limit peak vessel pressure during overpressure transients to satisfy ASME code requirements. The postulated transients for which safety/relief valve actuation is required are given in Chapter 14 and in Appendices R and Q of the Final Safety Analysis Report (FSAR). The automatic depressurization system provides a means to rapidly depressurize the primary system down to a pressure at which low-pressure cooling systems can provide makeup. In the event of a small or medium break Loss of Coolant Accident (LOCA), this function would be required if the High Pressure Coolant Injection (HPCI) system is unable to maintain vessel water level.

The most likely leakage paths through the Target Rock Corporation (TRC) two-stage safety relief valve are: (1) through the main stage, past the main disc and seat interface, or (2) through the pilot stage, past the disc and seat interface.

Representatives of General Electric (GE) and TRC (the valve's manufacturer) have in the past indicated that main stage leakage is typically substantial and increases faster than pilot stage leakage, and that pilot stage leakage is more common than main disc leakage.

History

SRV-3A experienced high discharge pipe temperatures during cycle 9, which required a similar evaluation (Reference 7). The following is a summary of events detailing unusual pilot leakage, which was eventually attributed to poor fitting insulation.

The SRV-3A pilot assembly was replaced during refueling outage (RFO) 8 in1991. The pilot assembly was replaced again on December 14, 1992 and September 2, 1993. Following the third replacement; the discharge pipe temperature on SRV-3A increased to 216°F on 9/28/93 and stayed at that temperature until 9/29/93 (less than 24 hrs) when the valve reseated and the discharge pipe temperature returned to drywell ambient. During the next incident; the discharge pipe temperature increased to about 235°F on 10/16/93 and returned to ambient on 10/20/93 when the reactor pressure / power was reduced due to malfunctioning of a main condenser circulating water pump.

SRV-3B had high discharge pipe temperatures between 1/24/96 and 2/6/96. The operability evaluation for that incident determined that the elevated temperature, which eventually settled out at 217 °F, was the result of minor pilot leakage (Reference 11 ). This was later confirmed at Wyle Labs where diagnostic "as-found " set point and leakage tests were performed.

SRV-3B leaked again in December of 1997 and stayed in service until April of 1999. While this trend data can be utilized for long term leakage analysis the as found set point data was distorted by corrosion by-products introduced during flood-up operations during the RFO.

SRV-3B developed a leak in October of 2000 (Reference 15). NRC approval for operability and continued operation was received on 2/21/01 (Reference16). The valve continued to leak until April of 2001 when it was removed for testing during the refueling outage. Special handling steps were incorporated to eliminate corrosion product contamination in an effort to improve as-found tests results. The as-found set point test for this valve was1127 psig or approximately 1% above set point. In addition, as found leakage was quantified to validate the measurement method used to quantify discharge pipe leakage.

Correlation of Current Conditions to Historical Experience

Due to the similarity of the increase in discharge pipe temperatures to previous leaking pilots the most probable cause for the leakage presently experienced by SRV-3D and SRV-3A is pilot leakage. This condition may clear with a lowering of reactor pressure. However, it will most likely remain in the 215 to 230 °F ranges from now until the next refueling outage.

The consequences of leakage across either the pilot or main stage boundary for SRV-3D or SRV-3A must be addressed, since leakage increases may occur later and may occur at either location. Pilot stage leakage affects valve lift set point and response time while main stage leakage does not. In either case neither the reactor dome safety limit (1325 psig), ASME code allowable for Upset conditions (1375 psig), or ASME code allowable for Emergency conditions (1500 psig) will be exceeded due to the SRV leakage bounded by this evaluation.

Pilot Stage Leakage

Pilot stage leakage can affect the performance of the two stage Target Rock SRV in the pressure-actuated mode (i.e., safety mode). The effects of leakage on valve performance have been extensively studied and consist of set point drift and response time changes (Reference 3).

The leakage rates studied by GE and TRC range from 200 lbs/hr to 1000 lbs/hr. Test results indicate that set point pressure increased by approximately 1% at a leakage rate of approximately 225 lbs/hr and by 2% at a leakage rate of approximately 400 lbs/hr. The set point then decreased 2% per 100 lbs/hr of additional leakage. The effect of leakage rate on set point is illustrated in Reference 3. Based on TRC test results, pilot stage leakage up to 1000 lbs/hr does not significantly affect the SRV set point (Reference 3).

Response time is the interval from pilot actuation to main disc lift. The normal response time for a two stage TRC SRV is approximately 0.4 seconds. Response time varies with leakage rate. A slower response time results in a higher peak reactor vessel pressure during the safety mode, and a faster response time results in a lower peak reactor pressure. A slower response time results when discharge pipe temperature increases. The impact of leakage on response time is presented in the "Impact on Nuclear Safety" section of this report.

Main Stage Leakage

Main stage leakage is an uncommon problem in the industry according to Target Rock. This is substantiated by information available on relief valve leakage, most of which is a result of pilot stage leakage. Leakage across the main stage boundary is an economic concern because of the potential for seat and/or disc damage. TRC and GE advise that leakage across the main disc will not affect the ability of the SRV to operate in either the pressure actuated, or power actuated modes. Leakage across the main stage should not cause the SRV to inadvertently open and cause a rapid depressurization or fail to reclose after operating.

Impact on Nuclear Safety

Neither the reactor dome safety limit of 1325 psig or the ASME code allowable of 1375 psig for peak vessel pressure are exceeded following a MSIV closure with flux scram. This conclusion takes into consideration the effects of the following variables that affect the results from the baseline analysis performed for the current cycle.

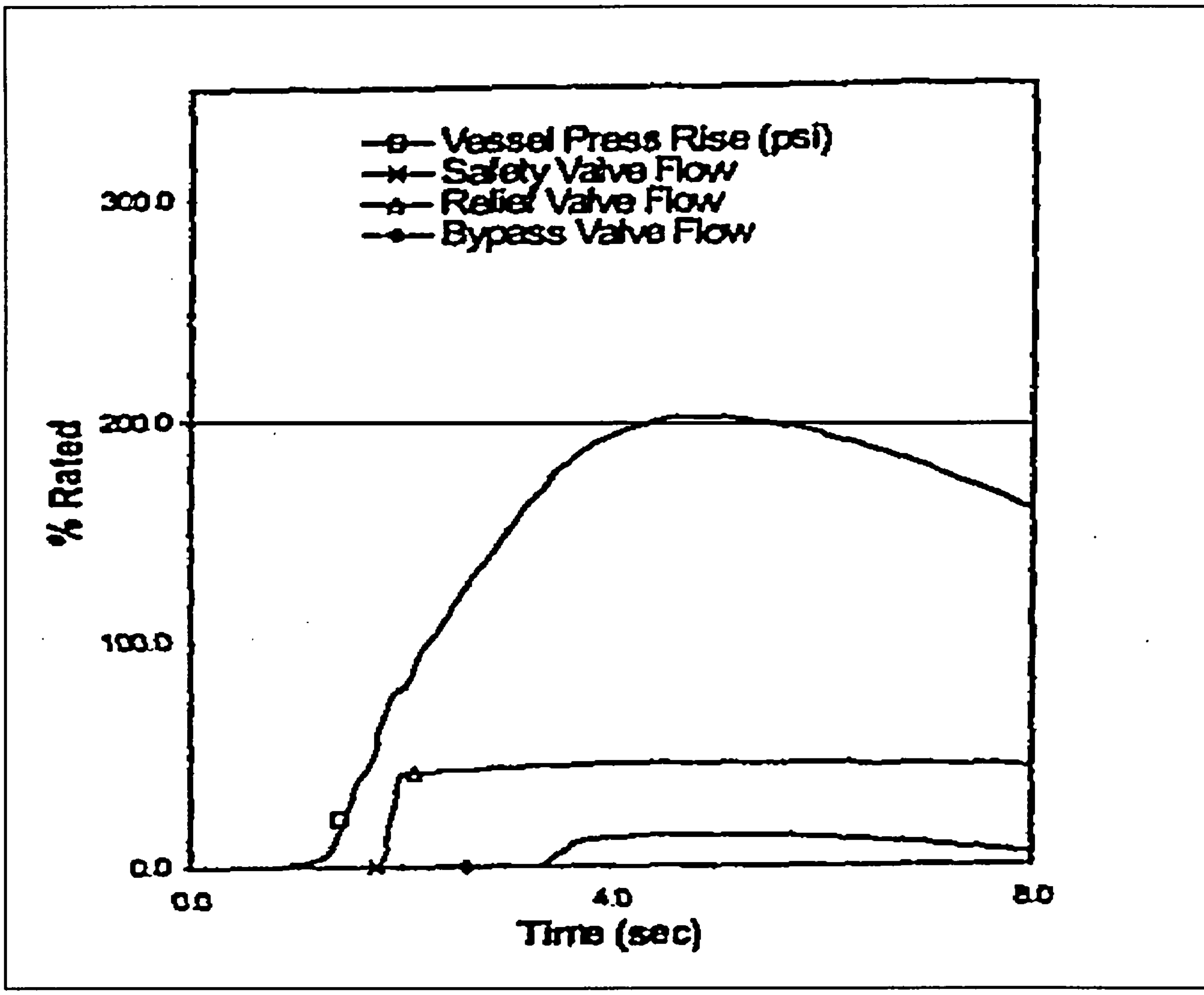
- The combined effect of a 10% increase in the nominal setpoint on all four SRVs is 30 psig (Reference 3).

- The combined effect of a response time delay of 0.9 seconds on two SRVs is 10 psig (Reference 3).

- The calculated peak vessel pressure of 1305 psig for the Cycle 15 MSIV closure with flux scram event did not take credit for the currently installed SRVs that have a 5.125 inch diameter throat. Instead the SRV throat sizes used were a range between 4.905 to 5.030 inches (Reference 15). The SRVs modeled in the Cycle 15 analysis (smaller throats) have a combined average rated steam flow of ~800,000 lbm/hr versus 870,000 lbm/hr for the larger throat SRVs now installed. The smaller throat sizes used in the Cycle 15 reload licensing analysis represent conservative inputs that increased the predicted peak reactor pressure during pressurization transients. The larger throat size actually installed provides greater than a 7% increase in relief capacity and will lower the predicted peak reactor pressure during pressurization transients.

The larger throat SRVs will reduce the peak vessel pressure provided the valves are near full open prior to the point in time that the peak vessel pressure is reached. The following Figure 32 from the MSIV Closure (Flux Scram) transient analysis results shows that all four SRVs are near full open at time (t) = ~2 seconds. The peak vessel pressure is reached at t = ~5 seconds (Reference 9). Therefore, the larger throat SRVs currently installed, but not modeled or reflected in the Cycle 15 MSIV Closure (Flux Scram) results, will effectively dampen the vessel pressure response. This conservatism is estimated to reduce the maximum pressure by approximately 5 psig.

**Excerpt from SRLR for PNPS Cycle 15**
**Figure 32**
**Plant Response to MSIV Closure (Flux Scram)**




Notes:

1. The initial dome pressure used in the analysis shown in Figure 32 was 1085 psig that corresponds to the high-pressure scram analytical limit.
2. In this figure the Y-axis is a dual scale. The Y-axis is in the units of psi for the vessel pressure rise data and % Rated for valve flow. 0.0 on the Y-axis in this figure corresponds to the initial peak vessel pressure of 1103 psig as tabulated below.

The peak vessel pressure result of 1305 psig is based on the following tabulation:

| | |
|---|---|
| 1085 psig | Initial dome pressure (Reference 9) |
| + 18 psig | Difference between peak vessel and dome pressure |
| 1103 psig | Initial peak vessel pressure |
| +202 psig | Vessel pressure rise during transient (Figure 32) |
| 1305 psig | Peak vessel pressure Cycle 15 MSIV Closure (Flux Scram) (References 9 & 17) |

The effects from SRV pilot leakage and throat size changes described above are combined as tabulated below to develop a conservative estimate of the peak vessel and dome pressures for the limiting MSIV Closure (Flux Scram) event:

| | |
|---|---|
| 1305 psig | Peak vessel pressure for Cycle 15, MSIV Closure (Flux Scram) assuming a 1% increase in nominal set point on all four SRVs and small SRV throat sizes used in previous cycles (Reference 9 & 17) |
| + 10 psig | Opening delay of 0.9 seconds for 2 SRVs (Reference 3) |
| + 30 psig | 10% increase in nominal setpoint of 4 SRVs (Reference 3) |
| (-) 5 psig | Est. peak pressure reduction from large SRV throat size |
| 1340 psig | Peak vessel pressure $\leq$ 1250 x 1.1 = 1375 psig |
| (-)18 psig | Difference between peak vessel and dome pressures |
| 1322 psig | Vessel dome pressure $\leq$ TS Safety Limit of 1325 psig |

The peak vessel pressure is estimated at 1340 psig which is significantly below the ASME code allowable of 1375 psig which is allowed to exceed the vessel design pressure of 1250 psig by 10% (i.e., 1.1 x 1250 psig = 1375 psig). The corresponding peak dome pressure of 1322 psig is less than the TS Safety Limit of 1325 psig.

The installed relief and spring safety valves protect the reactor coolant pressure boundary from exceeding the ASME Level C limit of 1500 psig during a full power ATWS. The limiting ATWS is a pressure regulator failure (PRFO) that causes the turbine control and bypass valves to fail open, leading to vessel depressurization and MSIV closure on low steamline pressure. This event was re-analyzed for the recent Thermal Power Optimization uprate from 1998 MWth to 2028 MWth as documented in Reference 18. This reanalysis credited the installed throat sizes of 5.125 inches and assumed one SRV experienced a 21 psig setpoint drift (equivalent to 1.8%) that caused the valve to open at 1136 psig (1115 psig + 21 psi drift). The remaining three SRVs were assumed to open at 1126 psig (1115 psig + 1% (11 psi)). The calculated peak vessel pressure for the limiting ATWS is 1495 psig which meets the acceptance criteria of 1500 psig. The selected discharge pipe temperature limit is chosen to limit setpoint drift to less than 1%, and therefore, the ATWS analysis described above bounds the conditions permitted by this operability evaluation and ensures the ASME Level C limit is not exceeded during the worst case ATWS.

The impact of either a delay in SRV response time or an increase in SRV opening pressure on critical power thermal margin is minimal. This is due to the rapid insertion of large negative control reactivity during transients before the higher pressure can contribute to any significant additional core power production due to core void collapse. This was demonstrated in NEDO-22159 (Reference 13), where a 30 psig increase in SRV opening set point resulted in only 0.1% increase in peak fuel rod heat flux following a limiting pressurization event. This was specifically evaluated for PNPS for cycle 6. It also applies to the current cycle due to the insignificant contribution of SRV pressure relief to the mitigation of the core power excursion associated with the limiting pressurization events on which the MCPR operating limit is based. Reactivity shutdown via reactor scram renders the core essentially sub critical before SRV pressure relief can be effective in moderating the void collapse due to the pressurization event.

The effect of the leakage on Torus water temperature is expected to be insignificant because the evaluated leakage is so small relative to the mass of the Torus water volume. Similarly, any effects on Drywell air temperature or containment pressure are also expected to be insignificant. No other systems are expected to be significantly impacted by this amount of leakage.

Each safety relief valve has one solenoid valve, which is attached to a manifold mounted on the air operator for the valve. The leakage flow through the safety relief valve will raise the temperature of the main valve body, base, pilot assembly and associated discharge pipe. The solenoid valve is environmentally qualified, which considers in part the normal ambient temperature to which it is exposed. The solenoid valve is not in direct contact with any part of the safety relief valve, which will experience appreciable elevated temperature because of the leakage through the valve. Therefore, the solenoid valve will not be exposed to any significant amount of conducted heat, but could be exposed to a slightly higher ambient temperature. The solenoid valve is mounted as an appendage off the safety relief valve in a configuration that maximizes air circulation around it, and minimizes the ambient temperature to which the solenoid valve is exposed. Therefore, the effects of minor leakage through the SRV-3D and 3A safety relief valves, is judged to have no appreciable affect on the environmental qualification of the safety relief valve solenoid.

SRV Leakage Versus Discharge pipe Temperature and SRV Set point

An SRV discharge pipe temperature of approximately 255°F can be correlated to a steam leakage flow rate of approximately 225 lbs/hr, while steam leakage of 1000 lbs/hr can be correlated to a discharge pipe temperature of approximately 275°F. It is acceptable to continue operation with a discharge pipe temperature of less than or equal to 255°F since test data has demonstrated that the possible relief valve set point drift at this temperature is equivalent to +1% (Reference 3).

Plant Parameter Effects on Discharge Pipe Temperature

Drywell Temperature: Sensitivity analysis predicts that the discharge pipe temperature is relatively insensitive to drywell temperature variations over the entire range of steam leakage (Reference 2).

Reactor Pressure: The temperature of the steam at the exit of the relief valve decreases as reactor pressure increases. Any effect on downstream discharge pipe temperature may be offset by increased leakage rates at higher reactor pressure. The temperature limit of 255°F was based on normal reactor operating pressure for the exit steam (Reference 2).

Containment Pressure: The safety relief valve discharge pipe is equipped with vacuum breakers that prevent drawing a column of torus water into the discharge pipe. The discharge pipe is at atmospheric pressure prior to inerting and slightly above atmospheric pressure after inerting the containment. The effects of containment pressure on discharge pipe temperature are negligible because the difference in discharge pipe pressure due to inerting is only a few psig. A leakage flow rate of up to 1000 lbs/hr will not be sufficient to pressurize the discharge pipe, thereby not affecting discharge pipe temperature (Reference 2). Therefore, containment pressure effects are judged to be negligible.

Conclusion

SRV-3D and SRV-3A are operable in the present condition. The leakage that has occurred is minor in nature and it is attributed to pilot stage. The present leakage levels are acceptable as discussed previously. Either intermittent or continuous leakage for either valve within the limits is acceptable for continued operation. Tests and analyses have shown that leakage rates of approximately 225 lbs/hr (equivalent to 255°F) should not impact the SRV set point by more than +1%.

4. Actions

Based on past experience with leaking pilot valves, a lower more conservative action limit has been selected for SRVs-3D and 3A in order to assure reliable operation and reduce damage to the pilot seat and disc. Therefore, if the discharge pipe temperature for SRV-3A or 3D exceeds 235°F for a period of 24 hours an orderly shutdown of the reactor shall commence and reactor pressure shall be less than 104 psig within 24 hours.

In addition, an increase in discharge pipe temperature to greater than 250°F may be an indication of a condition not previously observed which may place valve performance outside the bounds of this evaluation. If the discharge pipe temperature for either SRV exceeds 250° F then an orderly shutdown shall commence and the reactor pressure shall be less than 104 psig within 24 hours.

Technical Specification surveillance 4.6.D.3 requires that SRV discharge pipe temperature is logged daily. This surveillance shall be performed at an increased frequency of once per hour, to compensate for the reduced margin between the normal maximum discharge pipe temperature of 212°F and 235°F.

## 4. REGULATORY ANALYSIS

### 4.1 No Significant Hazards Consideration

Entergy Nuclear Operations, Inc. (Entergy) is requesting NRC approval of its evaluation of the acceptability of reactor operation at Pilgrim Nuclear Power Station for greater than 90 days with leakage past Safety Relief Valves 3A and 3D as indicated by discharge pipe temperatures in excess of 212 °F. This approval is requested in accordance with Technical Specification (TS) 3.6.D.4. Entergy has evaluated whether or not the requested approval involves a significant hazards consideration by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   Indication of elevated Safety Relief Valve (SRV) discharge pipes temperature is attributed to leakage past the SRVs. Excessive leakage, corresponding to temperatures greater than 255°F, has the potential to affect SRV operability by affecting the SRV setpoint or response time. Continued operation with the discharge pipes of the SRVs indicating temperatures less than 255°F ensures that the leakage past the SRVs is maintained below the threshold for a leakage rate that would potentially have an effect on SRV setpoint or response time.

Administrative controls are in place to ensure that margin to the 255°F value is maintained to assure reliable operation and to reduce the potential for damage to the pilot seat and disc. The SRVs continue to perform their intended design/safety function with no adverse effect because the leakage past the SRVs is maintained below the threshold for a leakage rate that could potentially have an adverse impact on the ability of the SRVs to perform their design functions. The impact of the leakage on other systems is small and all systems continue to be able to perform their intended design functions. Current accident analyses remain bounding and there is no significant increase in the consequences of any accident previously evaluated. In addition, as a result of the leakage, normal plant operating parameters are not affected and consequently there is no increased risk in a plant transient.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   Continued plant operation with elevated discharge pipe temperatures for SRV-3A & 3D within the bounds of the established administrative controls ensures that the leakage past the SRVs is maintained below the threshold for a leakage rate that would potentially have an effect on SRV setpoint or response time. This ensures that the SRVs will perform their intended design/safety function. The leakage does not adversely impact the ability of any system to perform its design function. The methods governing plant operation and testing remain consistent with current safety analysis assumptions. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

   Continued operation with the discharge pipes of SRV-3A & 3D indicating temperatures in excess of 212°F does not adversely affect existing plant safety margins or the reliability of the equipment assumed to operate in the safety analysis. The leakage does not result in excess SRV setpoint drift or response time changes. The imposed administrative controls on plant operation provide assurance that there will be no adverse effect on the ability of the SRVs to perform their intended design/safety function. There are no changes being made to safety analysis assumptions, safety limits or safety system settings that would adversely affect plant safety. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, Pilgrim concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

## 5. Environmental Consideration

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. The proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the proposed amendment.

## 6. REFERENCES

1. Technical Specifications and associated bases 3.6.D.1, 3.6.D.2, 3.6.D.3, 3.6.D.4 and 3.6.D.5.

2. General Electric Report NSE 13-0282 "Pilgrim Plant, SRV Tailpipe Steam Temperature Correlation for SRV Leakage Monitoring System" dated February 1982.

3. General Electric Report NEDE-30476, "Set point Drift Investigation of Target Rock Two-Stage Safety/Relief Valve (Final Report)", dated February 1984.

4. Operability Evaluation for Target Rock Corporation Two-Stage Safety Relief Valve 203-3D dated 8/17/1991.

5. NRC approval of operability evaluation for 203-3D, incoming NRC letter 1.91.288 dated 10/24/91.

6. TCH92-133, "Root Cause/Corrective Action Response for 203-3D" (PR92.0338/F&MR 91-373).

7. Operability Evaluation for Target Rock Corporation Two-Stage Safety Relief Valve 203-3A dated 11/2/93.

8. Wyle Lab. Test Report No. 41211-0 dated 4/25/1991.

9. Supplemental Reload Licensing Report for PNPS Reload 14, Cycle 15, GNF Report No. 0000-0008-6613SRLR, Rev. 1, March 2003.

10. MR #03101431, MR #03101432.

11. Operability Evaluation for Target Rock Corporation Two-Stage Safety Relief Valve 203-3B pilot serial number 1025 dated 2/9/96.

12. Operability Evaluation for Target Rock Corporation Two Stage Safety Relief Valve 203-3D pilot serial number 1054 dated 11/4/97.

13. NEDO-22159 General Electric Boiling Water Reactor Increased SRV Simmer Margin Analysis for PNPS Unit1-June 1982.

14. Operability Evaluation for Target Rock Corporation Two-stage Safety Relief Valve 203-3B EE98-004, 01/98.

15. Operability Evaluation for Target Rock Corporation Two Stage Safety Relief Valve 203-3B EE00-060 rev1 12/5/00.

16. NRC approval of Operability Evaluation for 203-3B, incoming NRC letter 1.01.011 dated 2/21/01 (TAC No. MB0874).

17. Calculation S&SA-174, "Reload Licensing Analysis Inputs – OPL-3 for Reload 14 Cycle 15".

18. GENE-0000-0000-6653, "Project Task Report T0902, 'Anticipated Transients Without Scram", January 2002, Rev. 0.

ENCLOSURE 2

Commitments
(1 page)

It is Entergy's commitment to enforce the following limits on continued operation with SRV discharge pipe temperatures:

| NUMBER | REGULATORY COMMITMENTS | DUE DATE |
|---|---|---|
| 1 | If the discharge pipe temperature for either SRV-3A or SRV-3D exceeds 235°F for 24 hours an orderly shutdown of the reactor shall commence and the reactor pressure shall be less than 104 psig within 24 hours. In addition, if the discharge pipe temperature for either SRV exceeds 250°F, an orderly shutdown of the reactor shall commence and the reactor pressure shall be less than 104 psig within 24 hours. | Whenever SRV-3A or 3D are above 212° F |
| 2 | Technical Specification surveillance 4.6.D.3 requires that SRV discharge pipe temperature be logged daily. This surveillance shall be performed at an increased frequency of once per hour. | Whenever SRV-3A or 3D are above 212° F |