Entergy

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

William J. Riggs
Director, Nuclear Assessment

October 24, 2003

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555-0001

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

Pilgrim Fourth Ten-Year Inservice Testing (IST) Program
Response to NRC Request for Additional Information

LETTER NUMBER: 2.03.122

REFERENCE: Entergy Letter No. 2.02.109, Pilgrim Fourth Ten-Year Inservice Testing (IST) Program and Request for Approval of IST Relief Requests, dated December 6, 2002.

Dear Sir or Madam:

The attachment to this letter provides Pilgrim response to NRC Request for Additional Information related to the IST relief requests discussed during a telephone call on September 8, 2003.

Based on this discussion, Entergy has taken the following actions:

- Pilgrim withdraws relief requests PR-02 and VR-05. These are no longer necessary to comply with IST requirements.

- Pilgrim also withdraws PR-04. In lieu of PR-04, Pilgrim requests a schedule relief up to June 2004, to procure the required instrumentation and implement the Code required testing as explained in response to NRC question No. 4.

- Pilgrim has revised the following relief requests incorporating additional information. See enclosure for NRC review and approval.

  PR-01
  VR-01, VR-02, VR-03, VR-04, and VR-06.

203122

A047

Upon receipt of approved relief requests, Pilgrim will docket the final IST Procedure No. 8.I.1.1.

If you have any questions or require additional information, please contact Mr. Bryan Ford, Licensing Manager, at (508) 830-8403.

Sincerely,

W. J. Riggs

Attachment: Pilgrim Response to NRC Request for Additional Information (4 pages)

Enclosures: Pump Relief Request, PR-01 (3 pages)
Valve Relief Request, VR-01 (3 pages)
Valve Relief Request, VR-02 (3 pages)
Valve Relief Request, VR-03 (7 pages)
Valve Relief Request, VR-04 (3 pages)
Valve Relief Request, VR-06 (6 pages)

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. Nuclear Regulatory Commission
Region 1
475 Allendale Road
King of Prussia, PA 19406

Senior Resident Inspector
Pilgrim Nuclear Power Station

ATTACHMENT

Pilgrim Response to NRC Request for Additional Information (RAI)
Fourth 10-year Interval Inservice Testing (IST) Program

Reference: Letter from Entergy Nuclear Operation, Inc, to NRC, "Pilgrim Nuclear Power Station Fourth 10-year Inservice Testing (IST) Program," dated December 6, 2002.

NRC Question 1:

General: The Pilgrim IST program must meet the requirements of the 1995 Edition including the 1996 Addenda of the ASME OM Code for pump and valve inservice testing for its fourth 10-year IST interval program pursuant to 10 CFR 50.55a(f)(4)(ii). In general, the staff finds the IST program was developed to meet the requirements in the 1995 Edition through 1996 Addenda of ASME OM Code although, in a few cases, the IST program cites the 1998 Edition through 2000 Addenda of the OM Code. Pursuant to 10 CFR 50.55a(f)(4)(iv), the IST program may meet subsequent editions and addenda that are incorporated by reference in 10 CFR 50.55a(b). Portions of editions and addenda may be used provided related requirements are met. The licensee is requested to clarify which portions of the later Code edition and addenda it intends to adopt including all related requirements and to document these later portions in the IST program.

Response:

The NRC has recently incorporated the 1998 Edition through 2000 Addenda of the OM Code for Operation and Maintenance of Nuclear Power Plants, (OMb Code - 2000) into the Code of Federal Regulations. The following portions of the OMb Code will be adopted into the revised IST Program. Pilgrim's review shows that there are no other related requirements, within the OMb Code – 2000, for these paragraphs. Pilgrim requests NRC approval to use the below identified portions of OMb Code-2000.

- Appendix I-1390, Test Frequency, Class 2 and Class 3 Pressure Relief Devices that are Used for Thermal Relief Application.
- Appendix I-4110(h) and Appendix I-4130(g), Pressure Relief Devices - a minimum of 5-minute time elapse between successive openings.
- Deletion ISTA 2.1, Inspection – duties of inspector, inspector qualifications, and access for inspector.

The previously submitted IST Program (Entergy Letter No. 2.02.109, dated December 6, 2002) section 4.0, Compliance paragraph 3 identifies the three revised Code sections that have been adopted.

NRC Question 2:

Relief Request PR-01: (a) A similar relief request was granted by the NRC in a previous 10-year IST program for Pilgrim pursuant to CFR 50.55a(f)(6)(i) on May 13, 1993. As part of the basis for granting the relief request, the licensee was requested to perform a study of the maintenance history of these pumps to determine if they are subject to frequent failures—especially where the degradation was not detected by the proposed alternative. However, in

the current relief request, the licensee did not discuss any details and/or record of maintenance for these pumps. The licensee is requested to discuss the maintenance history of these pumps including any recent failures.
(b) Also, the licensee has changed the relief request completely from an individual pump test to a combined pump test of both pumps. The licensee is requested to address the feasibility of using portable flow meters or instruments to measure the individual pump's flow and compare each pump's flow every 3 months. In addition, the licensee is requested to address the associated hardships (e. g. costs, design changes) necessary to modify the system such that both pumps can be tested independently.

Response:

(a): The relief request PR-01 has been revised and now includes maintenance history information. See the enclosure.

(b): The original relief request stated in general terms that the use of portable ultrasonic flow meters is not practical. The relief PR-01 has been revised to provide additional information about flow instrumentation feasibility and includes the associated hardships (design changes and costs). See enclosure.

NRC Question 3:

Relief Request PR-02: Please provide the drawings of all the major pump components (for the high pressure and booster pumps including the driver) as well as the combined pump assembly.

Response:

Entergy withdraws this relief request.

NRC Question 4:

Relief Request PR-04: In the section entitled Basis for Relief, the licensee states that meeting the procurement and calibration requirements for these instruments to cover the range to the lower extreme (2.3 Hz) is impractical due to the limited number of vendors supplying such equipment. The availability of these instruments might have been impractical a decade ago. However, it appears that these instruments are readily available from several vendors today. Please discuss your reasons for determining the impracticality of meeting this Code requirement today.

Response:

Pilgrim has located a vendor that can provide accelerometers that meet the Code requirements for the SBLC pump testing. As such, Pilgrim withdraws this relief request (PR-04) and requests a schedule relief up to June 2004 to procure the required accelometers and to implement the necessary controls for Code required testing.

NRC Question 5:

Relief Request VR-01:
(a) Please provide the system names in which excess flow check valves 1-CK-17A/B/C/D, 1-CK-18A/B/C/D, 12-CK-360, and 12-CK-361 are installed.

(b) Please provide the check valves numbers, which are installed in the referenced system "Reactor Water Cleanup System (1201)."

(c) Please provide the classification (i.e. Class A or C or AC) of all the check valves included in the relief request.

(d) In its Basis for Relief, the licensee states in the data was obtained from the period between 1983 and 1999 (RFO#12). Please provide available data from more recent periods between 1999 and 2003.

Response:

(a): The check valve information has been incorporated into the enclosed revised VR-01.

(b): The check valve numbers installed in the referenced system are: 12-CK-360 and 361.

(c): The enclosed revised relief request states the classification as AC (for gross leakage check).

(d): Pilgrim has incorporated recent information into the enclosed revised VR-01.

NRC Question 6:

Relief Request VR-02: In its Basis for Relief, the licensee states, "history search at PNPS shows zero failures (failure to close) have been observed from 1989 through 1999 RFO#12." Please provide available data from more recent periods between 1999 and 2003.

Response:

Pilgrim has incorporated the most recent information into the enclosed revised VR-02.

NRC Question 7:

Relief Request VR-05:

(a) The wording in relief request VR-05 under "Test Requirement" paragraph ISTC 4.2.8 (d) is not consistent with the wording in paragraph ISTC 4.2.8 (d) of the ASME OM Code Edition 1995 including 1996 Addenda. Please correct this discrepancy.

(b) Identify where, in the PNPS Technical Specification, the requirement for stroke-time testing (between 3 and 5 seconds) as indicated in the relief request is located.

(c) Under "Alternate Testing," the licensee states that the Code-referenced, stroke-time test will not be used. Paragraph ISTC 3.3 of the Code requires the establishment of reference stroke time. The licensee has not requested relief from ISTC 3.3. Therefore, the licensee must include in its request for relief an additional citation of paragraph ISTC 3.3 to be included under "Relief Requested."

(d) The purpose of the inservice testing program is to ensure operational readiness of components, whereas the purpose of Technical Specification requirements is to assess

system operability. The inservice testing of the components is required to meet the ASME OM Code requirements. The ASME OM Code is continually updated and enhanced based on industry experience and comments to Code committee. Please provide your reasoning for replacing Code requirements with Technical Specifications, and how these Technical Specification requirements will ensure component operational readiness.

Response:

Pilgrim withdraws relief request VR-05

NRC Question 8:

Relief Requests VR-03 and VR-06, VR-04
NRC will discuss its concerns with these relief requests during a conference phone call with the licensee.

Response:

Based upon the information provided by the NRC staff during the conference call on September 8, 2003, Pilgrim has revised these relief requests.

VR-03: The requested information related to the maintenance and/or failure history has been included into the enclosed revised relief request.

VR-04: This relief request is revised to clarify that Maintenance Rule activities performed to meet 10 CFR 50.65 are not intended to be part of the basis consideration. Also, additional clarifications have been added. The revised relief request is enclosed for NRC review and approval.

VR-06: The maintenance and/or failure history within the original relief request has been updated and expanded. All of the valves within this relief request now have 16 years of seat leakage test history. Also, VR-06 has been revised to better describe how this relief incorporates risk reduction features through the use of a Performance Based Test Program minimum threshold requirement, and grouping the pressure isolation valves as a penetration (pressure isolation boundary) pair. Revised VR-06 is enclosed.

SUMMARY:

PR-01 is revised (enclosed).
PR-02 is withdrawn.
PR-04 is withdrawn
VR-01 is revised (enclosed).
VR-02 is revised (enclosed).
VR-03 is revised (enclosed).
VR-04 is revised (enclosed).
VR-05 is withdrawn.
VR-06 is revised (enclosed)

ENCLOSURE

REVISED RELIEF REQUESTS

Pump Relief Request, PR-01 (3 pages)
Valve Relief Request, VR-01 (3 pages)
Valve Relief Request, VR-02 (3 pages)
Valve Relief Request, VR-03 (7 pages)
Valve Relief Request, VR-04 (3 pages)
Valve Relief Request, VR-06 (6 pages)

PUMP RELIEF REQUEST PR-01

3 pages

PUMPS: P-202A, P-202B, P-202C, P-202D, P-202E, P-202F

SYSTEM: Reactor Building Closed Cooling Water (RBCCW)

CLASS: 3

FUNCTION: Emergency Equipment Cooling

TEST REQUIREMENTS: ISTB 5.2.1, Group A Test. Group A tests shall be conducted with the pump operating at a specified reference point. The test parameters shown in Table ISTB 4.1-1 shall be determined and recorded as required by this paragraph. The test shall be conducted as follows:

ISTB 5.2.1(b): For centrifugal and vertical line shaft pumps, the resistance of the system shall be varied until the flow rate equals the reference point. The differential pressure shall then be determined and compared to its reference value. Alternatively, the flow rate shall be varied until the differential pressure equals the reference point and the flow rate determined and compared to the reference flow rate value.

RELIEF REQUESTED: Obtain normal inservice test parameters using single pump operation. When plant operating conditions do not allow scheduled single pump operation, perform the quarterly testing with two pumps running in parallel at the specified reference flow rate.

BASIS FOR RELIEF: Reactor Building Closed Cooling Water System instrumentation is not configured to measure individual pump flow rates during plant parallel pump operations in a single loop. Redesign of the system would be necessary to reconfigure the piping to allow the installation of permanent flow instrumentation or to utilize portable flow instrumentation.

The RBCCW piping configuration does not permit installation of flow instrumentation (either portable ultrasonic or standard flow orifices) on the pump discharge piping that would be consistent with good instrument practices. A flow measurement instrument that compromises the recommended industry installation practices will not provide accurate reliable flow results, on a consistent bases, to meet

the rigorous criteria for Code pump testing. Each RBCCW loop has three pumps. The common suction header splits into three separate pump suction lines, which supply each RBCCW pump and then return into a common discharge header. Adequate distance downstream of each individual pump discharge elbow is not available where discharge piping joins a common header.

Because the general area around the RBCCW pumps is very congested, the most cost effective fix requires a plant design change that will add a testing loop for both the 'A' loop and 'B' loop RBCCW systems. The modification would cut into each pump discharge header and run three test lines which connect into a common test loop that ties back into the 14" common pump discharge header. The estimated cost for design and engineering, materials, installation, and testing to implement this modification is $600,000 per RBCCW Loop (Total System cost of 1.2 million dollars).

A maintenance history study was performed on the RBCCW pumps to verify that they are not subject to frequent failures – especially where the degradation might not be detected by the proposed alternative testing. The study reviewed RBCCW pump historical data back to 1986 and confirmed that the RBCCW pumps (and motors) are not subject to frequent failures. The most common maintenance work activities have been periodic corrective measures to fix pump packing leakage and preventative maintenance for motor oil changes and pump coupling lubrication. There was one functional failure of the motor for P-202E within the study time period. This was a recent event that resulted from motor winding degradation. The motor windings failed while the P-202E was in service. The follow-up investigation revealed that the 30 year old motor windings had degraded due to age. The P-202E motor winding failure was not detectable (due to it's nature) through normal Code pump testing, in that, all pump and motor bearing vibration parameters remained in their normal band and pump hydraulic capacity remained unchanged.

PUMP RELIEF REQUEST PR-01 (CONTINUED)

The Reactor Building Closed Cooling Water System is part of the ultimate heat sink for containment cooling functions and Reactor Vessel shutdown cooling. Test loops do not exist for individual pump flow tests; therefore, disturbance of the system normal configuration during operation (and some cold shutdown conditions) will have a negative impact on the plant's ability to maintain safe steady-state operation.

The Reactor Building Closed Cooling Water pump system flow rates depend on reactor power, service water injection temperature, outside ambient temperature, and plant equipment heat loads. When plant service water heat loads are high and/or salt service water inlet temperature is high (during summer months), parallel pump operation is necessary. These conditions normally occur during the months of July, August, and September. Because of these limitations, the RBCCW Group A pump testing will be performed using one of two methods. Scheduled testing during these conditions will be conducted in accordance with ISTB 5.3, Pumps in Regular Use. The Group A testing using parallel pumps will be performed using the same pump configurations (i.e., P-202A/B, P-202A/C, P-202D/E, or P-202D/F) at a known reference point of total flow. The use of parallel pumps allows monitoring both of the pumps' hydraulic and mechanical (vibration) parameters for degradation.

During periods when plant heat loads and climatic conditions allow the scheduled quarterly testing to be performed using single RBCCW pump operation, the Group A Test Method will be conducted using a single pump at a known flow reference point.

ALTERNATE TESTING: Perform normal quarterly Type A pump testing using single RBCCW pump operation. When plant heat loads and climatic conditions do not allow single pump quarterly testing to be scheduled in advance, the Type A test will be performed while parallel pumps are operating at the specified reference flow rate.

VALVE RELIEF REQUEST VR-01 3 pages

SYSTEMS: Core Spray System (1400)
High Pressure Coolant Injection System (2301)
Reactor Core Isolation Cooling System (1301)
Reactor Water Cleanup System (1201)
Recirculation Pump Instrumentation (262)
Nuclear Boiler Instrumentation (261)
Nuclear Boiler Instrumentation (263)
Main Steam (1)

VALVES: Excess Flow Check Valves manufactured by Chemequip:

1-CK-17A/B/C/D (Sys 1)	262-26A/B	263-79
1-CK-18A/B/C/D (Sys 1)	263-38	263-81
12-CK-360 (Sys 1201)	263-44	263-83
12-CK-361 (Sys 1201)	263-45	263-90
1400-31A/B	263-51	263-92
1301-15A/B	263-53	263-215A/B
2301-26	263-55	263-217A/B
2301-220	263-57	263-219A/B
261-19A/B	263-59	263-220A/B
261-20A/B	263-61	263-223A/B
261-21A/B	263-69	263-225
261-22A/B	263-71	263-227
261-67A/B/C/D/E/F/G/H	263-73	263-231A/B
261-110A/B	263-75	263-233
262-25A/B	263-77	263-237
		263-242A/B

CATEGORY: AC

CLASS: 1, 2

FUNCTION: Excess Flow Check Valves (EFCVs) are installed within each instrument process line that is part of the Reactor coolant pressure boundary and that penetrates primary containment. Each EFCV closes to limit flow within the respective instrument line in the event of an instrument line break downstream of the EFCV.

TEST REQUIREMENT: ASME OM Code Subsection ISTC Paragraph 4.5.1 requires these valves to be tested nominally every 3 months, except as specified by Paragraph ISTC 4.5.2. The PNPS IST Program takes exception to the testing requirements in accordance with subparagraphs ISTC 4.5.2(c) and ISTC 4.5.2 (f).

ISTC 4.5.2(c): If exercising is not practicable during plant operation and cold shutdowns, it shall be performed during refueling outages.

ISTC 4.5.2(f): All valve testing required to be performed during a refueling outage shall be completed before returning the plant to operation.

ISTC 4.3.3(a) Frequency. Tests shall be conducted at least once every 2 years.

<u>RELIEF REQUESTED</u>: Relaxation of the number of EFCVs tested every refuel outage from "each" to a "representative sample" every refuel outage (nominally once every 24 months). The representative sample is based on approximately 20 percent of the valves each 2-year cycle such that each valve is tested at least every 10 years (nominal).

<u>BASIS FOR RELIEF:</u> NEDO-32977-A and the associated NRC Safety Evaluation, dated March 14, 2000, provide the basis for this relief. NEDO-32977-A justifies relaxing the EFCV testing frequency from the current testing of each valve once/cycle to an approximately 20% sample once/cycle such that each valve is tested within a 10-year interval.

NEDO-32977-A demonstrates, through operating experience, a high degree of reliability with EFCVs and the low consequences of an EFCV failure. Reliability data in the report (Tables 4-1 and 4-2) documents two EFCV failures (failure to close) at four participating plants (Monticello, Dresden, Vermont Yankee, and Oyster Creek) for Chemequip valves similar to those used at PNPS. These two failures were observed over a service time of 5426 operating years (4.75E+07 operating hours). This results in a "Best Estimate Failure Rate" of 4.21E-08 per hour of operating time and an "Upper Limit Failure Rate" of 1.33E-07 per hour of operating time. A review of historical test surveillance data and a test failure component history search at PNPS shows zero EFCV failures (failure to close or gross leakage test) have been observed based on data from 1983 through 2003 (RFO #14). In addition, there are no known EFCV failures that occurred earlier than 1983.

The instrument lines at PNPS have a flow-restricting orifice upstream of the EFCVs to limit Reactor water leakage in the event of rupture. Previous evaluations contained in PNPS's Updated Final Safety Analysis Report (FSAR) of such an instrument line rupture do not credit the EFCVs for isolating the rupture. Thus, a failure of an EFCV, though not expected as a result of this request, is bounded by the analysis. Based on NEDO-32977-A and the analysis contained in PNPS's FSAR, the proposed alternative to the required exercise testing frequency for EFCVs prescribed by OM-10 provides a satisfactory level of quality and safety.

ALTERNATE TESTING: This Relief Request proposes to exercise close test, by full-stroke to the position required to fulfill its function, a representative sample of EFCVs every refueling outage. During the close test, gross valve seat leakage (LEF) will be measured. The representative sample is based on approximately 20 percent of the valves each cycle such that each valve is tested every 10 years (nominal). An exercise open test will be performed on each valve following the exercise close test and leak testing.

EFCV failures will be documented in PNPS's Corrective Action Program as a surveillance test failure. The failure will be evaluated and corrected. The Administrative EFCV Sample Test Program procedure will trend EFCV test failures and determine whether additional testing is warranted.

The Administrative EFCV Sample Test Program procedure will also establish a minimum acceptance criteria for Chemequip EFCVs of less than or equal to one failure per year (two failures per 2 years) on a 2-year rolling average. This requirement will ensure EFCV performance remains consistent with the extended test interval. Upon exceeding the criteria, an evaluation will be required which will:

- require a root cause evaluation to determine cause;
- determine the extent of conditions;
- require an evaluation of the testing interval to ensure reliability of the EFCVs; and
- produce a risk analysis of the effects of the failures on cumulative and instantaneous plant safety.

Corrective actions and performance goals will be established based on the results of the root cause analysis.

REFERENCES: NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation," dated June 2000

Safety Evaluation by the Office of Nuclear Reactor Regulation related to a Relief Request for excess flow check valve testing frequency at Pilgrim Nuclear Power Station Docket No. 50-293, dated September 17, 2002 (TAC No. MB5122)

VALVE RELIEF REQUEST VR-02 3 pages

SYSTEM: Nuclear Boiler Instrumentation (263)

VALVES: 2-CK-125A, 2-CK-125B (manufactured by Dragon)

CATEGORY: AC

CLASS: 2

FUNCTION: Excess Flow Check Valves (EFCVs) are installed within each instrument process line that is part of the Reactor coolant pressure boundary (RCPB) and that penetrates primary containment. Each EFCV closes to limit flow within the respective instrument line in the event of an instrument line break downstream of the EFCV.

TEST REQUIREMENT: ASME OM Code Subsection ISTC paragraph 4.5.1 requires these valves to be tested nominally every 3 months, except as specified by ISTC paragraph 4.5.2. The PNPS IST Program takes exception to the testing requirements in accordance with subparagraphs ISTC 4.5.2(c) and ISTC 4.5.2(f).

ISTC 4.5.2(c): If exercising is not practicable during plant operation and cold shutdowns, it shall be performed during refueling outages.

ISTC 4.5.2(f): All valve testing required to be performed during a refueling outage shall be completed before returning the plant to operation.

ISTC 4.3.3(a) Frequency. Tests shall be conducted at least once every 2 years.

RELIEF REQUESTED: Relaxation of the number of EFCVs tested every refuel outage from "each" to a "representative sample" every refuel outage (nominally once every 24 months). The representative sample is based on approximately 20 percent of the valves (for the case of Dragon EFCVs, one valve) each 2-year cycle such that each valve is tested at least every 10 years.

BASIS FOR RELIEF: NEDO-32977-A and the associated NRC Safety Evaluation, dated March 14, 2000, provide the basis for this relief. NEDO-32977-A justifies relaxing the EFCV testing frequency from the current testing of each valve once/cycle to an approximately 20% sample once/cycle such that each valve is tested within a 10-year interval.

NEDO-32977-A demonstrates, through operating experience, a high degree of reliability with EFCVs and the low consequences of an EFCV failure. Reliability data in the report (Tables 4-1 and 4.2) documents two EFCV failures (failure to close) at three participating plants (Clinton, Fermi, and WNP2) for Dragon valves similar to those used at PNPS. These two failures were observed over a service time of 2494 operating years (2.18E+07 operating hours). This results in a "Best Estimate Failure Rate" of 9.2E-08 per hour of operating time and an "Upper Limit Failure Rate" of 2.89E-07 per hour of operating time.

A review of historical test surveillance data and a test failure component history search at PNPS show zero EFCV failures (failure to close) have been observed from 1989 through RFO 14 in 2003 (these values were initially installed in 1987, but could not undergo meaningful plant testing until 1989 because design actuation flow rate was greater than available system test flow rate).

The instrument lines at PNPS have a flow restricting orifice upstream of the EFCVs to limit Reactor water leakage in the event of rupture. Previous evaluations contained in PNPS's Updated Final Safety Analysis Report (FSAR) of such an instrument line rupture do not credit the EFCVs for isolating the rupture. Thus, a failure of an EFCV, though not expected as a result of this request, is bounded by the analysis. Based on NEDO-32977-A and the analysis contained in PNPS's FSAR, the proposed alternative to the required exercise testing frequency for EFCVs prescribed by OM-10 provides a satisfactory level of quality and safety.

ALTERNATE TESTING: This Relief Request proposes to exercise open test and exercise close test, by full-stroke to the position required to fulfill its function, a representative sample of EFCVs every refueling outage. During the close test, gross valve seat leakage (LEF) will be measured. The representative sample is based on approximately 20 percent of the valves (for the case of Dragon EFCVs, one valve) each cycle such that each valve is tested at least once every 10 years. An exercise open test will be performed on each valve following the exercise close test and leak testing.

EFCV failures will be documented in PNPS's Corrective Action Program as a surveillance test failure. The failure will be evaluated and corrected. The Administrative EFCV Sample Test Program procedure will trend EFCV test failure and determine whether additional testing is warranted.

The Administrative EFCV Sample Test Program procedure will also establish minimum acceptance criteria for Dragon EFCVs of less than or equal to 1/2 failure per year (one failure per 2 years) on a 2-year rolling average. This requirement will ensure EFCV performance remains consistent with the extended test interval. Upon exceeding the criteria, an evaluation will be required which will:

- require a root cause evaluation to determine cause;
- determine the extent of conditions;
- require an evaluation of the testing interval to ensure reliability of the EFCVs; and
- produce a risk analysis of the effects of the failures on cumulative and instantaneous plant safety.

Corrective actions and performance goals will be established based on the results of the root cause analysis.

REFERENCES:

NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation," dated June 2000

Safety Evaluation by the Office of Nuclear Reactor Regulation related to a Relief Request for excess flow check valve testing frequency at Pilgrim Nuclear Power Station Docket No. 50-293, dated May 2, 2001 (TAC No. MB1124)

VALVE RELIEF REQUEST VR-03

7 pages

SYSTEM: As Applicable

VALVES: All Category A and AC valves requiring periodic leakage rate testing with the exception of containment isolation valves, pressure isolation valves, and pressure-relief devices.

CATEGORY: A and AC

CLASS: 2

FUNCTION: Valves with seat leakage requirements; inventory preservation, intersystem leakage, and bypass flow. These valves are grouped by valve type, system application, and safety function and have been listed by their function 1) parallel pump bypass flow, 2) safety-related system "Q"-List/ seismic boundary integrity, and 3) boundary integrity.

TEST REQUIREMENT: ISTC 4.3.3(a), Frequency. Tests shall be conducted at least once every 2 years.

RELIEF REQUESTED: Seat leakage testing of the selected Category A and AC valves will be performed in accordance with the Performance-Based Testing (PBT) Program in lieu of Subsection ISTC 4.3.3(a), Frequency.

BASIS FOR RELIEF: A Performance-Based Testing Program (PBT Program) has been developed which relaxes the prescriptive OM Code seat leakage test frequency requirements and allows test intervals to be based on system service and component performance. Through a screening process, the PBT criteria will only be applied to valves that continue to exhibit a high degree of seat leakage reliability when the extended test interval is applied.

Through its own Regulatory Improvement Program, the NRC has instituted an ongoing effort to eliminate requirements that are marginal to improving safety, and to reduce the regulatory burden on utilities. A PBT Program, utilizing an extended testing interval based on the successful completion of two or more consecutive leakage rate tests, would take advantage of the findings of NUREG-1493 Appendix A. The conclusions drawn by the NUREG suggest that "if a component does not fail within two operating cycles, further failures appear to be governed by the random failure rate of the component". The NUREG also states that any test scheme considered should require a failed component to pass at least two consecutive tests before allowing the extended test interval to be applied.

VALVE RELIEF REQUEST VR-03 (CONTINUED)

The PBT Program for valves that require seat leakage testing under the OM Code, Subsection ISTC was developed in much the same manner as the Option B Program for Appendix J tested valves, which was permitted by amendment of the Code of Federal Regulations on October 26, 1995. In the studies performed in support of the Code change, it was concluded that performance-based testing is feasible without significant risk (NUREG 1493). Also, EPRI Research Project Report TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals", reaffirmed this position by stating that changes in leakage testing frequencies are feasible without significant risk impact.

The PNPS PBT Program utilizes over a decade of valve seat leakage test data for all the valves included within this program. A test history review was conducted for each valve to determine long-term valve performance and to obtain performance insights that can be used to screen out "suspect valves" that are either more prone to seat leakage failure or exhibit erratic seat leakage behavior. Valves that screened out, as "suspect valves" during the initial screening process have not been included within this relief request. This screening process uses Condition Monitoring methods to ensure that high degree valve seat leakage reliability is maintained. In this way, the PBT criteria will only be applied to valves that exhibit a high confidence level (through seat leakage trend data, maintenance history data, and valve failure analysis, when applicable) that seat leakage reliability will not be impacted when the extended test interval is applied.

All the valves selected for inclusion into the PBT Program were then categorized based upon their valve type (i.e., check, globe, gate, and ball) and system application to determine which specific valve groups may be more prone to failure. By grouping the valves, a comparison can be performed of like valves, which are in systems with similar service conditions to determine whether some valves, even though they have good seat leakage test histories, should remain on a 2-year test frequency.

Valves that exhibit normal operational behavior, pass a minimum of two consecutive tests, and have not been flagged as "suspect valves" will be placed on an extended test interval of 4 years or two refueling intervals, whichever is longer. Any valve not meeting the minimum threshold test performance requirement will be left on a 2-year test interval. When a seat leakage test failure occurs on any extended interval valve, the initial test frequency of 2 years must be re-established until an evaluation demonstrates that extending the test interval will not impact valve seat leakage performance and two consecutive seat leakage tests are acceptable.

ALTERNATE TESTING: Category A and AC valves that meet the PBT threshold criteria will have their seat leakage testing frequency controlled in accordance with the Performance-Based Testing Program. Valves that have met the threshold of passing two periodic consecutive tests will be permitted to be tested every 4 years or two refueling intervals, whichever is longer. Valves which fail their acceptance criterion will return to the 2-year test frequency. Before the extended test interval may be reinstated, an evaluation must be performed that demonstrates extending the test interval will not impact valve seat leakage performance, and the valve must pass a minimum of two consecutive seat leakage tests.

DISCUSSION: The following pages contain the PBT valve groups. The valves included within this Relief Request are categorized by the valve type, system application, and safety function requiring a seat leakage limit. Valves shall meet the applicable guidelines of the Nuclear Energy Institute (NEI-94-01) Industry Guidelines for Implementing Performance-Based Option of 10CFR Part 50, Appendix J for a performance-based testing program. Using the guidelines, valves that have passed a minimum of two consecutive leakage rate tests may be placed on an extended testing interval. All valves placed on an extended testing interval for seat leakage will still have all other associated ASME OMa Code testing (i.e., exercising and position verification) performed at the required frequency by the Inservice Testing Program. Valves that have not passed the minimum of two consecutive tests will continue to be tested once every two years until their test performance permits using the extended testing interval.

Each valve or combination of valves has been assigned an operational frequency rating, which is indicative of the expected frequency that the valve would perform an active function (i.e., opening and closing). The valve operational frequency when combined with system service conditions provides a useful indicator that provides insights related to the expected rate of valve degradation. The assigned operational frequency ratings are defined as follows: Seldom, Infrequent, Occasional, and Frequent.

Seldom - Maintenance or convenience type valves in which operation is seldom desired or required.

Infrequent - Valves in which operation would be expected at a cold shutdown or greater frequency for testing or other evolutions.

Occasional - Valves in which operation would be expected at a quarterly frequency for testing or other evolutions.

Frequent - Valves in which operation is expected during normal plant operation for reasons other than testing. Valves assigned as Frequent will be reviewed for exclusion from the performance-based testing program.

If the review shows that normal operation of a specific valve may impair long-term seat leakage reliability, then it will be excluded.

SAFETY RELATED SYSTEM "Q"-LIST/SEISMIC BOUNDARY INTEGRITY

Safety related systems that require an active/passive isolation between ASME Code Class and non-Seismic/non-"Q"-List boundary. The isolation ensures that safety-related systems carrying contaminated water post-accident will not leak outside the "Q"-List/Seismic boundary. These valves remain closed during normal plant operation and have their seat leakage limited to a specific maximum amount. Maintaining the seat leakage below a specified limit ensures that system boundary integrity.

Residual Heat Removal to Radwaste Crosstie - Occasional:

RHR System Discharge To Radwaste Flow Control Valve (MO-1001-21) and RHR System Discharge To Radwaste Block Valve (MO-1001-32) provide the boundary isolation between the RHR System and the Radwaste System. Maintaining seat leakage below the specified limit ensures the secondary containment bypass leakage is minimized. Seat leakage is currently being measured by the feed rate required to maintain test pressure in the test volume. A review of seat leakage testing data from 1987 to present shows that these valves have no test failures. There are no observed adverse seat leakage trends that point to degradation of the valve seating characteristics, nor is there erratic seat leakage test history.

Residual Heat Removal Keepfill to Condensate System - Occasional:

RHR A and B Keepfill Supply Check Valves (CK-1001-363A & 362B) provide the boundary integrity for the keepfill line. Maintaining seat leakage below the specified limit ensures the secondary containment bypass leakage is minimized. Seat leakage is currently being measured by either the feed rate required to maintain test pressure in the test volume or by measuring leakage through a downstream telltale connection while maintaining test pressure on one side. A review of seat leakage testing data from 1993 to present shows that these valves have no test failures. There are no observed adverse seat leakage trends that point to degradation of the valve seating characteristics, nor is there erratic seat leakage test history.

Core Spray to Condensate Storage Tank Crosstie - Seldom:

Core Spray Pump A and B Manual Suction Valves From Condensate Storage Tank (1400-2A & 2B) provide the boundary isolation between the Core Spray System and the Condensate Storage Tank. Maintaining seat leakage below the specified limit ensures the secondary containment bypass leakage is minimized. Seat leakage is currently being measured by either the feed rate required to maintain test pressure in the test volume or by measuring leakage through a downstream telltale connection while maintaining test pressure on one side. A review of seat leakage testing data from 1993 to present shows that these valves have no test failures. There are no observed adverse seat leakage trends that point to degradation of the valve seating characteristics, nor is there erratic seat leakage test history.

Core Spray Keepfill to Condensate System - Occasional:

Core Spray A and B Keepfill Supply Check Valves (CK-1400-212A & 212B) provide the boundary integrity for the keepfill line. Maintaining seat leakage below the specified limit ensures the secondary containment bypass leakage is minimized. Seat leakage is currently being satisfied by measuring leakage through a downstream telltale connection while maintaining test pressure on one side. A review of seat leakage testing data from 1993 to present shows that these valves have no test failures. There are no observed adverse seat leakage trends that point to degradation of the valve seating characteristics, nor is there erratic seat leakage test history.

HPCI Pump Suction From Condensate Storage Tank Crosstie - Occasional:

HPCI Pump Suction Valve From Condensate Storage Tank (CK-2301-20) provides the boundary isolation between the HPCI System and the Condensate Storage Tank. Maintaining seat leakage below the specified limit ensures the secondary containment bypass leakage is minimized. Seat leakage is currently being measured by the feed rate required to maintain test pressure in the test volume. A review of seat leakage testing data from 1993 to present shows that this valve has no test failures. There are no observed adverse seat leakage trends that point to degradation of the valve seating characteristics, nor is there erratic seat leakage test history.

RCIC Pump Suction From Condensate Storage Tank Crosstie - Occasional:

RCIC Pump Suction Valve From Condensate Storage Tank (CK-1301-23) provides the boundary isolation between the HPCI System and the Condensate Storage Tank. Maintaining seat leakage below the specified limit ensures the secondary containment bypass leakage is minimized. Seat leakage is currently being measured by the feed rate required to maintain test pressure in the test volume. A review of seat leakage testing data from 1993 to present shows that this valve has no test failures. There are no observed adverse seat leakage trends that point to degradation of the valve seating characteristics, nor is there erratic seat leakage test history.

SDV Vent and Drain Valves to Reactor Building Sump - Occasional:

Scram Discharge Volume Vent Valves (CV-302-21A/B & CV-302-23A/B) and Scram Discharge Volume Drain Valves (CV-302-22A/B & CV-302-24A/B) provide the boundary isolation between the SDV and the Reactor Building Sump. Maintaining seat leakage below the specified limit ensures the isolation from the Reactor coolant pressure boundary (RCPB) during a Scram is maintained. Seat leakage is currently being measured by the feed rate required to maintain test pressure in the test volume. A review of seat leakage testing data from 1991 to present shows that these valves have no test failures due to valve seat degradation. One post maintenance seat leakage test failure was observed following improper limit switch bracket adjustment. During the maintenance activity, improper alignment of a limit switch bracket prevented the valve from fully closing. The historical test data for all valves shows no observed adverse seat leakage trends that point to degradation of the valve seating characteristics, nor is there erratic seat leakage trend data for any of these valves.

PARALLEL PUMP BYPASS FLOW

Systems that require an active isolation of the parallel pump loop to perform the desired safety function may require the isolation feature to allow minimum system leakage.

Standby Liquid Control (SLC) System Parallel Pump Bypass Flow - Occasional:

SLC Pump A and B Discharge Check Valves (CK-1101-43A/B) allow flow of borated coolant to the Reactor Vessel upon activation and prevent pump bypass flow upon closure in the event of a pump's discharge relief failure to close tightly. Maintaining seat leakage below the specified limit ensures minimum pump bypass flow. Seat leakage is currently being measured by the feed rate required to maintain test pressure in the test volume. A review of seat leakage testing data from 1993 to present shows that these valves have no test failures. There are no observed adverse seat leakage trends that point to degradation of the valve seating characteristics, nor is there erratic seat leakage test history.

BOUNDARY INTEGRITY

Standby Liquid Control (SLC) Injection Line - Infrequent:

SLC Inboard Injection Check Valve (CK-1101-15) provides a pressure isolation barrier between the RCS from the high pressure portion of the SLC System. Normal penetration isolations include an inboard injection valve, an outboard injection valve, and squib valves. Upon firing of the squib valves with the RCS at operating pressure, this check valve becomes one of the two isolation barriers. Therefore, the valve performs an isolation barrier between the high pressure Reactor system to high pressure safety system interface. Seat leakage measurement is currently satisfied by collecting leakage at the upstream test connection while maintaining test pressure during the Reactor Pressure Vessel Leakage test. A review of seat leakage testing data from 1993 to present shows that this valve has no test failures. There are no observed adverse seat leakage trends that point to degradation of the valve seating characteristics, nor is there erratic seat leakage test history.

VALVE RELIEF REQUEST VR-04 3 pages

SYSTEM: As Applicable

VALVES: As Applicable

CATEGORY: A, B

CLASS: 1, 2, and 3

FUNCTION: Valves required providing remote position indication and allowing for proper operator action during normal operation, abnormal conditions, or emergency situations.

TEST REQUIREMENT: ISTC 4.1, Valve Position Verification. Valves with remote position indicators shall be observed locally at least once every 2 years to verify that valve operation is accurately indicated. Where practicable, this local observation should be supplemented by other indications such as use of flow meters or other suitable instrumentation to verify obturator position. These observations need not be concurrent. Where local observation is not possible, other indications shall be used for verification of valve operation.

RELIEF REQUESTED: Relief is requested from performance of the routinely scheduled position indication verification of at least once every 2 years.

BASIS FOR RELIEF: The Code position indication verification (PIV) requirement is not a true test, but rather a simple verification of valve indicator function. Therefore, the routinely scheduled (2 year frequency) position indication verification is not an activity that will detect valve degradation or provide added assurance of valve reliability. The routinely scheduled PIV is simply a reoccurring check that verifies the correlation between valve position and valve light indication. From valve reliability stand point, the post maintenance PIV (which is performed following maintenance work that has the potential to impact the relationship between valve stroke position and valve light indication) is the only PIV that has the ability to discover valve degradation (due to poor maintenance practices) or provide added assurance of valve reliability.

During the past 10 years (assisted by CFR50.65, Maintenance Rule efforts), plant maintenance controls have ensured that subsequent to work on safety related valves; proper post maintenance testing is performed prior to declaring the valve operable. The post maintenance testing controls for

valves with remote position indicators ensure that the remote valve position indicators accurately reflect valve travel direction. Work activities that require such a retest are controlled administratively by PNPS post maintenance guidelines. These strict procedural guidelines over maintenance (both corrective and preventive) activities ensure that proper checks and tests are performed.

A review of PNPS historical PIV data since 1987 (supports the proposed alternate testing) shows that no position indication failures have been detected during the Code routinely scheduled (2 year frequency) PIV activities during this time frame. There have been instances were valve maintenance was performed (incorrectly) in which the post maintenance valve position indication verification test has identified a (PIV) failure. Casual correspondence with other utilities has shown that plants with strict procedural controls over their maintenance activities have not experienced PIV failures during the routinely scheduled 2-year valve position indication verifications.

An Industry review has suggested that the primary means of identifying improper valve position is through system restoration (i.e., returning the system to it's normal operating or standby alignment) following a maintenance outage. Further review has shown that improper valve indication, except on rare occasions, is identified by the post maintenance testing activities, not by the routinely scheduled PIV activities.

The hardships encountered during the performance of the routinely scheduled 2-year periodic valve position verifications includes the following: There are substantial plant resources (Planning and Scheduling, Operations group, Instrumentation and Controls group, Programs Test group, Radiation Protection group, and ALARA group) dedicated to the performance of these verifications which require: prejob briefings (e.g., Control Room personnel must be assigned in lieu of other critical tasks), establishing effective communications (i.e., local and the Control Room), stationing personnel locally at the valve, and documentation of the verification results. Where local observation is not possible, other indications are used for the verification of valve operation, such as specialized tests (i.e., seat leakage, system flow, etc.).

Additionally, stationing personnel locally at a valve to enable observation during the valve stroke process to perform the PIV often requires admittance to radiation, high radiation, and locked high radiation areas. Since no PIV failures or valve degradation is being discovered during the routinely scheduled 2-year PIV's, performing the 2-year PIV undermines good surveillance ALARA practice.

The Code position indication requirement is not a test but a simple verification of indicator function. Therefore, the routinely scheduled 2-year PIV is not an activity that will detect degradation or provide added assurance of valve reliability, but simply verifies valve position. PNPS considers that the proposed alternate testing, and the continuation of current OM Code testing activities of switch-to-light stroke timing and full stroke exercising of active valves, provides good assurance that proper valve operation is accurately indicated. In addition, procedural controls for system restoration and post maintenance testing controls ensure that plant systems are properly aligned to accomplish their intended function.

<u>ALTERNATE TESTING</u>: Valve position indication verification (PIV) shall be performed only as a post maintenance testing activity.

VALVE RELIEF REQUEST VR-06

6 pages

SYSTEM: As Applicable

VALVES: All Category A and AC Pressure Isolation Valves (PIVs)

CATEGORY: A and AC

CLASS: 1

FUNCTION: Pressure Isolation Valves (PIVs) have seat leakage acceptance limits for providing Reactor Coolant Pressure Boundary (RCPB) isolation. PIVs are configured as two isolation valves in series providing redundant pressure isolation between the higher Reactor pressure and the lower rated pressure system.

TEST REQUIREMENT: ISTC 4.3.3(a) Frequency. Tests shall be conducted at least once every 2 years.

RELIEF REQUESTED: Pressure Isolation Valves will be placed into a Performance-Based Testing (PBT) Program in lieu of testing to Subsection ISTC 4.3.3(a), Frequency. The PIVs will be seat leak tested individually to monitor and assess leak tightness performance, but will be grouped as a penetration (pressure isolation boundary) pair to increase the total reliability of the penetration pressure isolation function.

BASIS FOR RELIEF: A Performance-Based Testing Program (PBT Program) has been developed which relaxes the prescriptive OM Code seat leakage test frequency requirements and allows test intervals to be based on system service and component performance. This PBT Program incorporates a risk reduction feature, which requires the grouping of PIV's as a penetration (pressure isolation boundary) pair to increase the total reliability of the penetration pressure isolation function.

Through its own Regulatory Improvement Program, the NRC has instituted an ongoing effort to eliminate requirements that are marginal to safety and to reduce the regulatory burden on utilities. A PBT Program, utilizing an extended testing interval based on the successful completion of two or more consecutive leakage rate tests, would take advantage of the findings of NUREG-1493 Appendix A. The conclusions drawn by the NUREG suggest that "if a component does not fail within two operating cycles, further failures appear to be governed by the random failure rate of the component". The NUREG also states that any test scheme considered should require a failed component to pass at least two consecutive tests before allowing the extended test interval to be applied.

The applicable penetrations and their associated pressure isolation valves have been assigned the highest quality group (Quality Group A) for design, fabrication, testing, and inservice inspection. These design and fabrication requirements minimize the probability of an accidental rupture of the penetration or those lines connected to the penetration. The testing and inservice inspection requirements ensure piping and components maintain their operability and structural integrity throughout the plant's service life. The design configuration (i.e., two valves in series) satisfies the need to protect against a single failure within the RCPB, as it pertains to an intersystem LOCA. When the frequency extension criteria are assigned to each penetration valve pair, the probability that a single failure will result in an event that has significant consequence is greatly reduced.

The PBT Program for PIVs that require seat leakage testing according to the OM Code was developed in much the same manner as the Option B Program for Appendix J tested valves, which was permitted by amendment of the Code of Federal Regulations on October 26, 1995. In the studies performed in support of the Code change, it was concluded that performance-based testing is feasible without significant risk (NUREG 1493). Also, EPRI Research Project Report TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals", reaffirmed this position by stating that changes in leakage testing frequencies are feasible without significant risk impact.

NUREG-1493 did <u>not</u> specifically address the impact on safety where Containment Isolation Valves (CIVs) also serve as PIVs. Regulatory Guide 1.163 endorsed the NEI guidance document but placed limitations on certain valves whose frequencies could not be extended unless the additional risk was considered and appropriately evaluated. The NRC has identified PIVs as fitting this group of valves requiring further risk justification. PNPS has reduced the overall risk of these dual-purpose valves (i.e., CIV/PIV) to an acceptable level of quality and safety by applying the PBT Program test frequency extension criteria to both valves in the PIV penetration pair (even though each valve is tested individually).

The three types of PIV penetration configurations at PNPS are listed below. Each configuration also specifies the associated valve's safety function for RCPB isolation (PIV) and/or primary containment isolation (CIV):

- <u>Configuration No. 1</u>: Inboard Check Valve (PIV), First Motor-Operated Valve (PIV/CIV) and Second Motor-Operated Valve (CIV). The penetration piping and components providing the RCPB isolation function are designed and tested to Quality Group A. The remaining outboard portion, designated for

containment isolation, is designed and tested to Quality Group B. The second motor-operated valve (non-PIV) is designed to allow closure at normal Reactor pressure, thereby providing a backup isolation feature.

- Configuration No. 2: Inboard Motor-Operated Valve (PIV/CIV) and Outboard Motor-Operated Valve (PIV/CIV). The penetration piping and components provide RCPB isolation and containment isolation and are designed and tested to Quality Group A. Both of these valves have automatic isolation interlocks, which initiate valve closure (and does not allow reopening using electrical power) when Reactor coolant pressure is greater than 70 psig. These valves cannot inadvertently be placed in the open position due to the aforementioned interlocks and the actuator is not designed with sufficient force to overcome the pressure differential across the disk.

- Configuration No. 3: Inboard Feedwater Check (CIV), Inboard HPCI System Check Valve (PIV), and Outboard Motor-Operated Valve (PIV/CIV). The penetration piping and components that provide both RCPB isolation and containment isolation are designed and tested to Quality Group A. These penetrations allow high pressure injection of makeup water into RCS; therefore, the system is designed for high pressures and equipped with additional isolation features.

When assessing the risk, it is important to note that each penetration configuration is designed and constructed to withstand the long-term high pressure service associated with the RCS out to the furthermost designated isolation valve from containment. For configurations 1 and 3, a degraded PIV can be provided with backup isolation capability by an additional leak tested isolation valve (CIV). Even though the CIV test is performed with air at relatively low pressure, the CIV test data does provide an important indicator about the valve's internal health. These valves are designated as CIVs only, but are designed to the same pressure/temperature ratings as Reactor Coolant System piping and components, therefore, are capable of performing an RCPB isolation function, if necessary. Also, the plant design provides instrumentation to monitor abnormal penetration leakage. Should leakage be detected, there is a very high probability that closing the additional high pressure isolation valve, through operator action, would provide an acceptable barrier between the attached downstream piping and the downstream low pressure components.

The PIV PBT Program utilizes over 16 years of valve seat leakage test data for each valve included. Hydrodynamic leakage testing for valves categorized as PIVs at PNPS began in 1987. Currently, nine satisfactory

periodic hydrodynamic leak rate tests, at the maximum function differential pressure (typically 1040 psig), have been completed for each PIV. Of the total PIV population at PNPS, only two valves (CK-1001-68B and CK-1301-50) have experienced test failures. A Design Change was implemented in 1999, which addressed the failure of CK-1001-68B by installing a swing check to replace the original tilting disk check. The failure mechanism was determined to be improper hinge pin alignment and excessive machining performed during corrective maintenance in 1987. The machining error resulted in removal of an excessive amount of base material from the valve seating area, which over the long term degraded this PIV's seat leakage reliability. The new replacement has undergone at least three successful periodic leakage tests performed in 1999, 2001 and 2003. The CK-1301-50, which was replaced with an upgraded valve in 1984, has experienced two seat leakage failures, one in 1989 and one in 1991. The failure mechanism was determined to be a design flaw associated with the manual exerciser internal mechanism, where as a square key was not properly secured in the manual exercising shaft keyway. The detachment of the key, and subsequent lodging of the key in the valve internal exerciser, resulted in the disk not making full contact with the valve seat. Corrective measures implemented in 1989 failed to provide a reliable fix for the key detachment problem. A 1991 modification was performed on the shaft keyway to key interface, which securely attaches the key in a most reliable fashion. Since the 1991 modification, there have been six successful PIV seat leakage tests.

A review of test history was conducted for each PIV to determine long-term valve performance and to obtain performance insights that can be used to screen out PIVs that are either more prone to failure or exhibit erratic behavior. A comparison was performed of like PIVs situated in systems with similar service conditions to determine whether certain valve models or types with good test histories should continue to be monitored on a two year test frequency because of an inherent design weakness or system application problem that could adversely impact seat leakage reliability. No significant issues that could impact seat leakage reliability for the PIVs within this relief request were identified.

A PIV penetration pair (share a common penetration) that passes a minimum of two consecutive tests, exhibits normal operational behavior, and have not been screened out as "suspect valves" will be placed on an extended test interval of 4 years or two refueling intervals, whichever is longer. Any valve pair not meeting the minimum PBT threshold requirement will be left on a 2-year test interval. In addition, if a failure occurs on any extended interval PIV, the initial test frequency of 2 years shall be re-established for both PIV's in the in the penetration pair. All test failures will be evaluated for cause and effect. Before the extended test interval may be reinstated, an evaluation must be performed that demonstrates extending the test interval will not impact valve seat

leakage reliability, and both PIVs in the penetration pair pass a minimum of two consecutive seat leakage tests.

ALTERNATE TESTING: Hydrodynamic leak rate testing of PIVs will be performed using the frequencies specified within the PIV Performance-Based Testing (PBT) Program. Penetration valve pairs that meet the threshold requirements for this PBT Program and pass two consecutive seat leakage tests, will be permitted (both valves in the pair) to be tested every 4 years or two refueling intervals, whichever is longer. A valve that fails its acceptance criterion shall require both of the PIVs in the associated penetration pair to be tested on a 2-year test frequency. When a seat leakage test failure occurs on any extended interval valve, the initial test frequency of 2 years must be re-established for both PIVs in the in the penetration pair. Before the extended test interval may be reinstated, an evaluation must be performed that demonstrates extending the test interval will not impact valve seat leakage reliability, and both PIVs in the penetration pair pass a minimum of two consecutive seat leakage tests.

DISCUSSION: The following is a listing of PIV pairs associated with a common penetration. Valves affected by this Relief Request perform the safety function of RCPB isolation and require a specified seat leakage limit. In addition the PBT threshold requirements identified within this relief request, these valves shall meet the applicable guidelines of the Nuclear Energy Institute (NEI-94-01) Industry Guidelines for Implementing Performance-Based Option of 10CFR Part 50, Appendix J for a performance-based testing program. Using these guidelines, a penetration valve pair that has passed a minimum of two consecutive leak rate tests may be placed on an extended testing interval. All valves placed on an extended testing interval for seat leakage will continue to receive any other associated OM Code required testing (i.e., exercising, stroke timing and position verification) at the IST Program specified frequencies. A valve pair, in a penetration, that has not passed the minimum of two consecutive tests will require both valves in the penetration to be tested during each refueling outage until their test performance permits an extended testing interval.

REACTOR COOLANT PRESSURE ISOLATION

A Pressure Isolation Valve (PIV) pair in a common penetration is defined as two normally closed valves in series that isolate the Reactor coolant system (RCS) from an attached low-pressure system. These valves are tested to minimize the probability of an intersystem LOCA and are normally operated (open and close) on an infrequent basis (cold shutdown or refueling outage). The infrequent service that each valve experiences, due to open and close cycling, has no impact on long-term seat leakage reliability.

Maintaining the seat leakage below the specified limit ensures the proper leak tightness of the RCS pressure boundary.

Residual Heat Removal (RHR) Injection/Suction Lines Penetration Valve Pairs:

- RHR Loop A Injection Line Penetration Valve Pair consists of CK-1001-68A, RHR Injection Line Check Vlv, and MO-1001-29A, LPCI Loop A Injection Valve #2 (Configuration No. 1).
- RHR Loop B Injection Line Penetration Valve Pair consists of CK-1001-68B, RHR Injection Line Check Valve, and MO-1001-29B, LPCI Loop B Injection Valve #2 (Configuration No. 1).
- RHR Shutdown Cooling Line Penetration Valve Pair consists of MO-1001-47, RHR Shutdown Cooling Outboard Isolation Valve, and MO-1001-50, RHR Shutdown Cooling Inboard Isolation Valve (Configuration No. 2).

Core Spray (CS) Injection Lines Penetration Valve Pairs:

- Core Spray Loop A Injection Line Penetration Valve Pair consists of CK-1400-9A, Core Spray Injection Check Valve, and MO-1400-25A, Core Spray Loop A Injection Valve #2 (Configuration No. 1).
- Core Spray Loop B Injection Line Penetration Valve Pair consists of CK-1400-9B, Core Spray Injection Check Valve, and MO-1400-25B, Core Spray Loop B Injection Valve #2 (Configuration No. 1).

High Pressure Core Injection (HPCI) Line Penetration Valve Pair:

- HPCI Injection Line Penetration Valve Pair consists of CK-2301-7, HPCI Discharge Check Valve, and MO-2301-8, HPCI Injection Valve #2 (Configuration No. 3).

Reactor Core Isolation Cooling (RCIC) Injection Line Penetration Valve Pair:

- RCIC Injection Line Penetration Valve Pair consists of CK-1301-50, RCIC Discharge Check Valve, and MO-1301-49, RCIC Pump Discharge Injection Valve #2 (Penetration Configuration No. 3).