August 14, 2001

Mr. Theodore Sullivan
Vice President - Operations
Entergy Nuclear Northeast
James A. FitzPatrick Nuclear Power Plant
Post Office Box 110
Lycoming, NY 13093

SUBJECT: FITZPATRICK - NRC INSPECTION REPORT 50-333/01-05

Dear Mr. Sullivan:

On June 30, 2001, the NRC completed an inspection at the James A. FitzPatrick Nuclear Power Plant. The results of this inspection were discussed on July 20, 2001, with you and members of your staff. The enclosed report presents the results of that inspection.

This inspection was an examination of activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. Within these areas, the inspection consisted of a selected examination of procedures and representative records, observations of activities, and interviews with personnel.

The NRC identified one finding regarding ineffective corrective actions for repetitive leakage past the main steam isolation valves that was evaluated under the risk significance determination process and was determined to be of very low safety significance (Green). This finding has been entered into your corrective action program and is discussed in the summary of findings and in the body of the attached inspection report. Furthermore, this finding was determined to involve a violation of NRC requirements, but because of the very low safety significance, this violation was non-cited.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room). Should you have any questions regarding this report, please contact me at 610-337-5211.

Sincerely,

/RA/

Glenn W. Meyer, Chief
Projects Branch 3
Division of Reactor Projects

Docket No. 50-333
License No.: DPR-59

Enclosure: Inspection Report 50-333/01-05

Attachment: Supplemental Information

cc w/encl:
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J. Knubel, VP Operations Support
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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.:	50-333
License No.:	DPR-59
Report No.:	50-333/01-05
Licensee:	Entergy Nuclear Northeast Post Office Box 110 Lycoming, NY 13093
Facility:	James A. FitzPatrick Nuclear Power Plant
Location:	268 Lake Road Scriba, New York 13093
Dates:	May 20 - June 30, 2001
Inspectors:	R. A. Rasmussen, Senior Resident Inspector B. J. McDermott, Senior Resident Inspector, VY T. A. Moslak, Health Physicist T. F. Burns, Reactor Inspector S. Dennis, Reactor Inspector
Approved by:	G. W. Meyer, Chief Projects Branch 3 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000333/01-05, on 05/20 - 06/30/2001; Entergy Nuclear Northeast, James A. FitzPatrick Nuclear Power Plant; Maintenance Rule Implementation.

The report covers a six-week inspection by resident inspectors, a baseline specialist inspection of radioactive material processing and transportation, and a specialist inspection of independent spent fuel storage installation (ISFSI) activities.

These inspections identified one Green issue that was a non-cited violation (NCV). The significance of most findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609 "Significance Determination Process" (SDP). Findings for which the SDP does not apply are indicated by "No Color" or by the severity level of the applicable violation. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website at http://www.nrc.gov/NRR/OVERSIGHT/index.html.

Barrier Integrity

GREEN. The inspectors determined that Entergy failed to take adequate corrective actions to prevent the local leak rate testing failures of main steam isolation valves for three consecutive operating cycles. These failures resulted in a recurring containment leakage pathway through the D main steam line.

The leakage path through the D main steam line was evaluated using the significance determination process and determined to be an issue of low safety significance (Green) based on an engineering analysis of the large early release frequency. This finding was a non-cited violation of NRC requirements. (Section 1R12.1)

TABLE OF CONTENTS

Report Details

SUMMARY OF PLANT STATUS

The reactor operated at full power for the majority of the inspection period.

1. REACTOR SAFETY

Initiating Events, Mitigating Systems, Barrier Integrity [REACTOR - R]

1R04 Equipment Alignments

a. Inspection Scope

During maintenance of the B emergency diesel generator (EDG) conducted the week of June 4, the inspectors performed walkdowns of the A and C EDG systems and the emergency service water (ESW) system. During this maintenance activity, ESW was partially isolated for replacement of the B EDG jacket water cooler.

The inspectors also performed a walkdown of the standby liquid control system following maintenance and testing during the week of May 28.

During these walkdowns the inspectors verified that significant valves and circuit breakers were in the appropriate position by comparing actual component position and the position described in the applicable operating procedures. The inspectors also performed visual inspections of the material condition of the major system components.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

The inspectors toured several plant areas and observed conditions related to fire protection. Inspectors looked for transient combustible materials; observed the condition of suppression systems, penetration seals, and ventilation system fire dampers; and verified fire doors were functional. These included:

- The EDG rooms.
- The reactor feed pump rooms.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation

.1 Repetitive Failures of Main Steam Isolation Valves

a. Inspection Scope

NRC Inspection Report 50-333/2000-12, Section 1R12, described the performance history of the main steam isolation valves (MSIVs) at FitzPatrick. This issue, which was determined to be a violation of NRC requirements, was left unresolved pending completion of the SDP risk assessment (URI 0500033/2000-12-001). The details of the risk assessment and a synopsis of the violation are provided below.

b. Findings

The inspectors determined that Entergy failed to take adequate corrective actions to prevent the local leak rate testing failures of main steam isolation valves for three consecutive operating cycles. These failures resulted in a recurring containment leakage pathway through the D main steam line. The leakage path through the D main steam line was evaluated using the SDP and determined to be an issue of low safety significance (Green) based on an engineering analysis of the large early release frequency. This issue was dispositioned as a non-cited violation of NRC requirements.

Background

For three consecutive refueling outages, the licensee reported via LERs that containment leakage rates exceeded authorized limits. In each case, the primary contributor to the containment leakage was the main steam isolation valves (MSIVs). Five out of eight MSIVs have failed the individual leakage requirements in each of the past three outages, with three main steam line valves failing in all three outages.

In each of the LERs, the licensee attributed the failure mechanism to normal wear and damage caused by valve cycling. Following the 1998 failures the licensee performed an evaluation and developed a modification package (JD-99-089) to improve valve performance. Key attributes of this modification included improved valve body and actuator guiding to reduce damage to the valve seat during cycling.

NRC previously reviewed the 1996 and 1998 failures, issued a non-cited violation in NRC Inspection Report 50-333/99-03 for the failure to meet technical specification requirements, and evaluated the proposed corrective actions. However, once the needed modifications were developed, the licensee deferred implementation of the modification until the next refueling outage. This decision did not adequately consider the performance trend and the need to continually comply with technical specifications.

The inspectors reviewed the performance history and evaluated the adequacy of the actions taken following the failures in 1998. The January 2000 modification package described the past history and why the modifications were necessary. In a section titled “Reason for Change,” the modification package stated, “Maintenance personnel have typically been repairing the seating surfaces until the valve passes the leak test. This type of approach does not correct the root cause of the seat leakage.” Based on the information in the modification package, the inspectors concluded that the repairs performed following the 1998 outage were similar to the repairs performed following the 1996 outage, which were ineffective in improving performance. Therefore, the inspectors concluded that the failures identified in the 2000 outage were expected and preventable. The inspectors concluded that the actions taken following the 1998 refueling outage were inadequate to prevent the failures in 2000.

Significance Determination

The SDP was applied to the leakage on the D main steam line containment penetration. The leak rate for the penetration exceeded both the capacity of the local leak rate testing (LLRT) equipment and the technical specification limit. Since this finding represented an actual open pathway in the physical integrity of the reactor containment, the Phase 1 SDP screen required a Phase 2 SDP evaluation.

The Phase 2 SDP process for containment barrier issues is conducted in accordance with the guidance provided in Inspection Manual Chapter (IMC) 0609, Appendix H. IMC 0609, Appendix H did not provide specific guidance on establishing the risk associated with penetration leakage for Mark I containments. Therefore, a Phase 3 risk evaluation was conducted to determine a change in the large early release frequency (LERF). The Phase 3 risk assessment uses the best available risk information to make a risk informed decision on the significance of inspection findings.

The Phase 3 risk evaluation used risk insights from Entergy’s modified level 2 PRA model, the independent plant examination (IPE), and various NUREGs and other published information regarding MSIV leakage in BWRs. Entergy modified their current level 2 PRA model to include MSIV leakage as a contributor to containment failure. Entergy determined that the change in LERF for the MSIV leakage identified was ~ 9.82E-08/year. This estimate was slightly less than the SDP Green/White threshold for change in LERF (1E-7/year). Therefore, based on this information alone, this finding would result in a very low safety significance (Green) finding.

Because the results of Entergy’s level 2 PRA analysis results were close to the green/white threshold, an independent bounding calculation was performed by the Region I Senior Reactor Analyst and the NRR PRA Branch Chief. This calculation used the conservative assumption that all sequences documented in the IPE that resulted in core damage at high pressure in the reactor vessel would result in a large early release. The results of this conservative analysis was a change in LERF of 1.5E-07/yr. Although this value is slightly above the green/white threshold, it confirmed that Entergy’s detailed level 2 PRA analysis result (9.82E-8/yr) was reasonable.

The NRC's Significance Determination Process and Enforcement Review Panel (SERP) considered several qualitative factors in reaching a final risk determination for this issue. Considerations which tended to decrease the risk associated with this finding were that 1) the leak testing methodology (lower pressure test and the inboard MSIV was tested in a direction opposite the actual flow direction following an accident) tended to overestimate the leak rate for MSIV penetrations; and 2) there would be significant deposition of radioactive nuclides in the main steam lines and condenser, thus limiting the radiological inventory transported to the site boundary and limiting the potential for a large early release. Factors considered that would tend to increase the risk estimate would be the addition of external initiating events (seismic, fire, etc) to be considered initiating events. The SERP determined that the mitigating factors would dominate the escalating factor in the change in LERF calculation. Therefore, the SERP determined that both the quantitative LERF calculation and qualitative factors supported the conclusion that this finding was of very low risk significance (Green).

Requirements

10CFR50, Appendix B, Criterion XVI, Corrective Actions, requires that equipment failures are corrected. Technical Specification (TS) Section 3.7.2 requires that primary containment integrity shall be maintained at all times when the reactor is critical. To verify containment integrity, TS Section 4.7.A.2.b, requires that leakage through each MSIV is less than or equal to 5.4 standard liters per minute (SLM) when tested at greater than or equal to 25 psig. Additionally, TS Section 6.20, specifies a maximum total pathway leakage of less than 320 SLM. Contrary to the above, following two consecutive failures of MSIV's 29AOV-80B, 29AOV-80D, and 29AOV 89D, the corrective actions were inadequate to prevent a subsequent failure. Each of these valves exceeded 5.4 SLM when tested individually, and the total pathway exceeded 320 SLM. This issue was evaluated using the SDP and was determined to be of low safety significance (Green). Therefore , this violation is being treated as a Non-Cited Violation, consistent with Section VI.A of the Enforcement Policy, issued on May 1, 2000 (65FR25368). The issues associated with this violation are in the Entergy corrective action system as DER 00-05158. **(NCV 05000333/2001-05-01)**

.2 Other Maintenance Rule Activities

a. Inspection Scope

The inspectors reviewed the June 17 failure of the A service water strainer. The failure was determined to be caused by a piece of wood debris that lodged in the strainer and prevented it from turning and washing. The inspectors reviewed the failure evaluation and corrective actions. The service water system was previously classified as (a)(1) by the Entergy Maintenance Rule program.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessment and Emergent Work

a. Inspection Scope

The inspector reviewed Entergy's assessment of plant risk due to planned maintenance on the high pressure coolant injection (HPCI) system during the week of February 10. NRC Inspection Procedure 71111, Attachment 13, and the following documents were referenced during this inspection:

- NUMARC 93-01, Revised Section 11, "Assessment of Risk Resulting from Performance of Maintenance Activities"
- Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants"
- Entergy's AP-10.02, "13-Week Rolling Schedule"
- Entergy's PDSO-09, "Use of Sentinel and Configuration Log"

Entergy's weekly risk assessment and assessment of emergent work on the A low pressure coolant injection (LPCI) system inverter were compared with the generic and FitzPatrick specific guidance documents. The inspector reviewed control room logs to confirm what actions were taken in response to the failure of A LPCI inverter on February 15, 2001.

The inspectors also reviewed the risk assessment during a period of potential grid instability caused by offsite testing by a nearby utility. In this case, Entergy was informed of the testing a day in advance and therefore did not have an opportunity to include the test in the normal work week review. The inspectors interviewed operators to determine the scope of review performed for the test. The inspectors then reviewed work in progress to assure compatibility.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the following operability determinations performed to address issues identified with safety significant systems. The inspectors reviewed the Final Safety Analysis Report (FSAR), and as applicable viewed the discrepant condition.

- A ground issue on the A and D APRM circuits.
- Unanticipated control rod block monitor alarms due to system software changes.
- A failure of the RCIC torus suction valve to open due to a broken wire in the valve control circuit.

- An analysis of EDG jacket water heat exchanges following the discovery that the heat exchangers had pit indications in excess of the latest wall thinning calculations.
- An evaluation of the EDG jacket water cooler expansion boots that had fasteners contacting the rubber portion of the expansion joint.
- An evaluation of the B EDG governor output shaft arm that failed during the EDG maintenance period.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications

a. Inspection Scope

The inspector reviewed design change JD-00-107, "Extended Load Line Limit Analysis (ELLLA) Changes," to evaluate how this change would affect the containment response following a loss of coolant accident. A report on the supporting analyses, JAF-RPT-MISC-04054, "FitzPatrick Operation Under Extended Load Line Limit Analysis (ELLLA) and Power Uprate," and nuclear safety evaluation 00-032, "Extended Load Line Limit Analysis (ELLLA) Implementation," were also reviewed. The inspector compared the revised peak containment pressure and the available margin, based on the design and licensing basis.

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing

a. Inspection Scope

The inspectors observed and reviewed the post maintenance testing associated with the following:

- Maintenance activities on the B EDG.
- Testing of the B and D APRM circuits following ground circuit modifications.

The inspectors reviewed technical specifications, the FSAR, and compared the testing requirements to those described by the site's administrative procedure for post maintenance testing. The inspectors verified that the testing met the appropriate test objectives.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors reviewed procedures and observed portions of testing related to the following surveillance tests:

- ST-3PB - Core Spray Loop “B” Quarterly Test
- ISP-100D-RPS - RPS Instrumentation Functional Test
- ST-8Q, Emergency Service Water System Quarterly Testing

The inspectors reviewed technical specifications, the FSAR, and system drawings. The inspectors verified that the testing met the appropriate test objectives. Additionally, for ST-3PB, Core Spray Loop “B” Test, the inspectors verified that the test was adequate following maintenance on the core spray B discharge flow indication switch (WR#01-00864-02).

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed the following temporary modifications (TMODs) to verify that the safety functions of associated systems were not affected, to assess the potential impact on control room operations, and for 10 CFR 50.59 applicability. The inspectors also performed a walkdown of the installed TMODs to ensure consistency with the TMOD documentation.

- 01-20 - Reactor total core flow indication - recorder problems
- 01-019 - Reactor feed pump instrumentation.

b. Findings

No findings of significance were identified.

E2 Engineering Support of Facilities and Equipment

E2.2 Independent Spent Fuel Storage Installation (ISFSI) Project (IP 60854)

a. Inspection Scope

Entergy was actively engaged in preparations for the dry run exercise of loading, closure (by welding), handling, unloading, and transfer of the HI-STORM 100 Cask System. The inspector performed an evaluation of pre-operational plans and testing activities for fuel load and cask sealing.

The inspector performed a walk-down of the cask transfer route. The route was examined to verify the pathway for cask movement was free of obstructions that might impede safe movement to, from, and within the reactor building spent fuel pool location.

The inspector examined the seal welding apparatus, reviewed the welding procedures and qualification records, and interviewed welding craftsmen performing welding on the full scale mock-ups of the multi-purpose canister shell, lid, and cover plates. The inspector examined completed weld test samples which had been removed from the full scale mock-ups and destructively tested to determine the effectiveness of the welding process. The inspector reviewed the certificate of compliance, evaluated current project staffing levels, personnel welding qualifications and training activities. Also, the inspector evaluated project staff awareness of industry operating experience.

b. Findings

No findings of significance were identified.

2 RADIATION SAFETY

Public Radiation Safety [PS]

2PS2 Radioactive Material Processing and Transportation (71122.02)

a. Inspection Scope

During the period May 21 - 25, 2001, the inspector conducted the following activities to verify that Entergy's radioactive material processing and transportation programs complied with the requirements of 10 CFR 20, 61, and 71 and Department of Transportation (DOT) regulations contained in 49 CFR 170-189.

The inspector conducted a walkdown, with the radwaste operations supervisor of liquid and solid radioactive waste processing systems to verify that the current system configuration and operation agree with the descriptions contained in the FSAR and the Process Control Plan. As part of this walkdown, the status of inactive radioactive waste processing equipment, including the waste concentrators, centrifuges, and the thin film evaporator was reviewed. A tour and independent radiological surveys were made of the Interim Waste Storage Facility to confirm the accuracy of material inventories and posted survey results, and that radioactive material containers were properly labeled.

The inspector reviewed the radio-chemical analysis results for each of Entergy's radioactive waste streams including dry active waste, spent bead and powdered resin, and irradiated hardware to determine if scaling factors for difficult-to-measure radio-nuclides were properly developed and applied in classifying the waste.

Five (non-excepted) radioactive material shipments were reviewed to determine that the packages complied with applicable NRC and DOT requirements. Included in this review were shipments of de-watered bead and powdered resin (Manifest Nos. 00-0448, 00-0449), mechanical filters and assorted dry active waste (Manifest No. 00-0451), and irradiated hardware (Manifest Nos. 00-0450, 00-452).

The inspector reviewed various Quality Assurance Department oversight reports that related to the implementation of the radwaste processing and transportation programs. Included in this review were audits (A00-09J and A00-13J) and field observation surveillances (2152, 2153, 2160, 2220, 2222). Problems identified from these oversight activities were confirmed to be in the corrective action program.

The following Radwaste Shipping Department self-assessments were reviewed to assess Entergy's effectiveness in evaluating program performance

- Loading 8-120B cask with spent fuel pool debris.
- Verification of transportation limits in shipping procedures.
- Dry active waste segregation.
- Review of DER's related to implementation of the radioactive material control program for adverse trends.
- Validation of the RADMAN 2000 software for designating radwaste packaging labels.

The inspector reviewed Deviation/Event Reports (DERs) relating to the control of radioactive material and work activities to determine if problems were identified in a timely manner and if appropriate actions were taken to evaluate and resolve the underlying issues. Included in this review were DER's Nos. 01-01839, 01-01223, 01-01855, 01-00514, 01-01387, 00-03200, and 00-01023.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems

Inspection findings presented in Section 1R12.1 of this report also had implications regarding Entergy's implementation of the corrective action system. As described above, the inadequate implementation of the corrective action program resulted in repetitive failures of the MSIVs. For the purpose of this inspection, this issue was dispositioned as an individual violation of 10 CFR 50, Appendix B, "Corrective Actions."

Additional items associated with the corrective action program were reviewed without findings.

4OA6 Meetings

.1 Exit Meeting Summary

On July 20, 2001, the inspectors presented the inspection results to Ted Sullivan and members of the Entergy staff. The inspectors asked whether any materials examined during the inspection should be considered proprietary. Where proprietary information was identified, it was returned to Entergy after review.

During the exit one finding of very low safety significance was discussed, which was determined to be a non-cited violation (NCV). Should Entergy elect to contest this NCV, a written response within 30 days of the date of this Inspection Report, with the basis for the denial, should be sent to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, and the NRC Resident Inspector at the FitzPatrick facility.

.2 Public Meeting

The NRC conducted the annual end of cycle review meeting with Entergy on June 19, 2001. During the meeting, the NRC discussed the status of the performance indicators, inspection findings, and performance trends for the past year. Entergy provided a brief synopsis of ongoing initiatives to address the areas of regulatory concern. The meeting was conducted in the FitzPatrick training center and was open for public observation. A copy of the slide presentation is available in ADAMS under Accession No. ML012180036.

ATTACHMENT 1

SUPPLEMENTAL INFORMATION

a. Key Points of Contact

J. Ratigan	Assistant Radiation Protection Manager
A. Zaremba	Director of Safety Assurance
D. Robert	Radwaste Operations Supervisor
G. Brownell	Licensing Engineer
G. Tasick	Licensing Manager
G. Thomas	Director Design Engineering
K. Pushee	Radiation Protection Manager
K. Phy	Dry Cask Storage Senior Project Manager
M. Colomb	Plant Manager
N. Starkweather	Journeyman Radiation Protection Technician
R. Phelps	Radwaste Shipping/Decontamination Supervisor
T. Sullivan	Site Executive Officer

b. List of Items Opened, Closed and Discussed

Opened and Closed

NCV 50-333/2001-005-001:	Failure to complete adequate corrective actions for leaking main steam isolation valves. (Section 1R12.1)

Closed

URI 05000333/2000-12-001:	Failure to take adequate corrective actions for leaking main steam isolation valves. (Section 1R12.1)

c. List of Acronyms

AP	Administrative Procedure
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
DER	Deficiency and Event Report
DOT	Department of Transportation
EDG	Emergency Diesel Generator
ELLLA	Extended Load Line Limit Analysis
ESW	Emergency Service Water
FSAR	Final Safety Analysis Report
HPCI	High Pressure Coolant Injection
IPE	Individual Plant Evaluation
IR	Inspection Report
ISFSI	Independent Spent Fuel Storage Installation
LER	Licensee Event Report
LERF	Large Early Release Frequency
LLRT	Local Leak Rate Test
LPCI	Low Pressure Coolant Injection
MSIV	Main Steam Isolation Valve
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
PARS	Publicly Available Records
QA	Quality Assurance
SDP	Significance Determination Process
SLM	Standard Liters per Minute
ST	Surveillance Test
TMOD	Temporary Modification
TS	Technical Specification
WR	Work Request

d. List of Documents Reviewed

RTID-98-016, Evaluation of Reasonableness of Using Composite Bead Resin Analysis Rather Than Smears as the Basis for Scaling Factors in accordance with 10 CFR 61
RP-OPS-05.08, CNS 1-13C, Cask Handling Procedure
RP-OPS-05.14, CNS 8-120B Cask Handling Procedure
FO-OP-032-41802, Setup and Operating Procedure for RDS-1000
RP-OPS-05.04, Radioactive Waste Data Base Control Program
RP-OPS-03.01, Radiological Survey Performance and Documentation
RP-OPS-05.06, Interim Waste Storage Facility
AP-07.07, Radioactive Waste Minimization Program
Self-Assessment: Loading 8-120B Cask with SFP Cleanup Debris
Self-Assessment: Verification of Transportation Limits in Shipping Procedures
Self-Assessment: Trash (DAW) segregation
Self-Assessment: Review DER's related to radioactive material control for adverse trends
Self-Assessment: Validate RADMAN 2000 software for designating radwaste packaging labels
QA Audit A00-09J, Process Control Program and Regulatory Guide 1.21
QA Audit A00-13J, Radiation Protection Program implementation during Refueling Outage 14
Manifest No.00-00-0448, Dewater Powdered Resin, Class B
Manifest No. 00-0449, Dewatered Bead/Powdered Resin, Class A
Manifest No. 00-0450, Irradiated Hardware, Class C
Manifest No. 00-0451, DAW, Filters, Class B
Manifest No. 00-0452, Irradiated Hardware, Class C
GWP-5, Rev 0, Spent Fuel Cask Welding
SS-8/8-HW, Rev 0, Welding Procedure Specification
PQR 2486, Rev 0, Procedure Qualification Record, GTAW Machine Welding
WQR 2662, Welder Qualification Record (Welder B-62)
DWG 1402, Rev 15, HI-STAR 100 MPC-68 Construction (Weld Details)
1014, Certificate of Compliance for Spent Fuel Storage Casks
CHO, Rev 0, Cask Handling Operations-Dry Run Plan