UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
SAM NUNN ATLANTA FEDERAL CENTER
61 FORSYTH STREET SW SUITE 23T85
ATLANTA, GEORGIA 30303-8931

January 28, 2002

Mr. Dale E. Young, Vice President
Crystal River Nuclear Plant (NA1B)
ATTN: Supervisor, Licensing &
Regulatory Programs
15760 West Power Line Street
Crystal River, FL 34428-6708

SUBJECT: CRYSTAL RIVER UNIT 3 - NRC INTEGRATED INSPECTION REPORT 50-302/01-04

Dear Mr. Young:

On December 29, 2001, the NRC completed an inspection at your Crystal River Unit 3. The enclosed report documents the inspection findings which were discussed on January 14, 2002, with you and members of your staff.

The inspection examined 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. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, the inspectors identified one issue of very low safety significance (Green).

Immediately following the terrorist attacks on the World Trade Center and the Pentagon, the NRC issued an advisory recommending that nuclear power plant licensees go to the highest level of security, and all promptly did so. With continued uncertainty about the possibility of additional terrorist activities, the nation's nuclear power plants remain at the highest level of security and the NRC continues to monitor the situation. This advisory was followed by additional advisories, and although the specific actions are not being released to the public, they generally include increased patrols, augmented security forces and capabilities, additional security posts, heightened coordination with law enforcement and military authorities, and more limited access of personnel and vehicles to the sites. The NRC has conducted various audits of the Florida Power Corporation's response to these advisories and Crystal River's ability to respond to terrorist attacks with the capabilities of the current design basis threat. From these audits, the NRC has concluded that the Crystal River security program is adequate at this time.

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/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Son Q. Ninh, Acting Chief
Reactor Projects Branch 3
Division of Reactor Projects

Docket No. 50-302
License No. DPR-72

Enclosure: Inspection Report 50-302/01-04

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U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket No:	50-302
License No:	DPR-72
Report No:	50-302/01-04
Licensee:	Florida Power Corporation (FPC)
Facility:	Crystal River Unit 3
Location:	15760 West Power Line Street Crystal River, FL 34428-6708
Dates:	September 30 - December 29, 2001
Inspectors:	S. Stewart, Senior Resident Inspector S. Sanchez, Resident Inspector G. Kuzo, Senior Radiation Protection Specialist (Sections 2OS1, 2OS2) J. Blake, Senior Project Manager (Section 1R08) W. Bearden, Reactor Inspector (Sections 1R02, 1R08, 1R17) G. Hopper, Operator License Examiner (Section 1R11.2) L. Mellen, Operator License Examiner (Section 1R11.2) M. Scott, Senior Reactor Inspector (Sections 1R02, 1R17) T. Morrissey, Resident inspector, Vogtle (Sections 1R02, 1R17)
Accompanied: Personnel	R. Chou, Reactor Inspector
Approved by:	Son Ninh, Acting Chief Reactor Projects Branch 3 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000302-01-04, on 09/30/2001 - 12/29/2001, Florida Power Corporation, Crystal River Unit 3, Event Followup and Refueling Outage.

The inspection was conducted by resident inspectors, two operator license examiners, a senior project manager, two reactor inspectors, and a regional health physics inspector. The inspection identified one Green finding. 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 web site.

A. Inspector Identified Findings

Cornerstone: Initiating Events

Green. The inspectors identified that the licensee did not consider worker fatigue in the licensee investigation of a potential loss of 4160 volt bus that involved worker performance issues.

This finding is more than a minor because it was viewed as a precursor to a significant event (loss of decay heat removal). The finding is considered to be of very low safety significance because no actual loss of equipment occurred. (Section 1R20)

B. Licensee Identified Violations

Violations of very low safety significance which were identified by the licensee have been reviewed by the inspectors. Corrective actions taken or planned by the licensee appeared reasonable. These violations are listed in Section 4OA7 of this report.

Report Details

Summary of Plant Status

Crystal River Unit 3 was shutdown for the planned 12R refueling outage until October 25, 2001. Full power operations were resumed on October 28, 2001. Power was reduced to 55 percent on December 14 for planned feedwater system maintenance. The reactor was returned to full power on December 16, 2001.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, [Reactor-R]; Emergency Preparedness [EP]

1R01 Adverse Weather Protection

a. Inspection Scope

The inspectors reviewed licensee Operating Instruction OI-13, Adverse Weather Conditions, Freezing Weather Preparations and Monitoring, to assure that measures were available to protect and monitor vital systems and components during cold weather periods. Final Safety Analysis Report, Chapter 2 was reviewed for design features associated with freezing weather mitigation. The emergency feedwater tank room was included in the site walkdown to verify that the cold weather mitigation strategies of sealing the room and installing portable space heaters could be implemented.

b. Findings

No findings of significance were identified.

1R02 Evaluations of Changes, Tests, or Experiments

a. Inspection Scope

The inspectors reviewed ten CFR50.59 safety evaluations, in the Initiating Events, Mitigating Systems, and Barrier Integrity cornerstone areas, to confirm that the license had appropriately reviewed and documented the changes in accordance with 10 CFR 50.59 and licensee procedures' CP-213, Revision 9, Preparation of a Safety Assessment and Unreviewed Safety Question Determination and REG-NGGC-001, 10 CFR 50.59 Reviews, Revision 2. Further, the inspectors considered the conditions under which changes to the facility or procedures may be made without NRC approval. The inspectors also reviewed 12 changes for which the licensee had determined that 10 CFR 50.59 evaluations were not required, to confirm that the licensee's conclusions to "screen out" these changes were correct and consistent with 10 CFR 50.59 and above procedures.

The major package related documents reviewed are listed in this report that included procedures, engineering calculations, modifications (MARs), work orders, site drawings, and corrective action documents (PCs, NTMs, NCRs, and CRs). The inspectors also reviewed additional information as necessary such as applicable sections of the Final

Safety Analysis Report (FSAR), the current FSAR update packages, the Crystal River design basis documentation, supporting analyses, technical specifications, and procedures.

In addition, the inspectors reviewed licensee audits and assessments to confirm that the licensee was identifying 10CFR50.59 issues, entering issues into the corrective action program, and was resolving the concerns.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

a. Inspection Scope

The inspectors conducted partial alignment walkdowns of risk important systems to evaluate the readiness of the redundant trains or backup systems while one train was out of service for maintenance. The walkdowns included switch and valve position checks looking for discrepancies with operating procedures in effect, and verification of electrical power to critical components. The inspector reviewed sections of the plant operating instructions as applicable to each walkdown. Nuclear condition reports were reviewed to verify that the licensee was identifying and correcting component alignment issues. The specific systems walked down were:

- Decay Heat Closed Cycle Cooling System, Train B, when the Train A heat exchanger was out of service for cleaning and preventive maintenance. The walkdown was conducted by verifying that critical Train B components and switches were in positions consistent with licensee drawing FD-302-631, Sheet 2, Decay Heat Closed Cycle Cooling and operating procedure OP-404, Decay Heat Removal System.

- Control Complex Chiller Train A when Train B was out of service for preventive maintenance. The walkdown was conducted by verifying that critical valves and control switches were in positions consistent with operating procedure OP-409, Plant Ventilation System.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

The inspectors conducted tours of risk significant plant areas to assure controls for transient combustibles and ignition sources were consistent with the licensee's Fire Protection Plan and 10 CFR Part 50, Appendix R. The inspectors also evaluated the material condition, operational lineup, and operational effectiveness of fire protection systems and assessed operational status and material condition of fire barriers used to contain fire damage using the standards of the Fire Protection Plan, 10 CFR Part 50, Appendix R, the Florida Power Corporation Analysis of Safe Shutdown Equipment, and the Final Safety Analysis Report. The inspectors reviewed sections of Administrative Instruction AI-2200, Guidelines for Handling, Use, and Control of Transient Combustibles and observed performance of SP-800, Monthly Fire Extinguisher Inspection and SP-802, Fire Hose Hydro Test and Hose Reel Inspection, to verify the operational condition of fire protection equipment. The components and areas receiving specific fire protection walkdowns were:

- B Emergency Diesel Generator Room
- Control Complex Ventilation Room
- Main Control Room
- Emergency Feedwater Pump Building

b. Findings

No findings of significance were identified.

1RO6 Flood Protection Measures

a. Inspection Scope

The inspectors reviewed the Final Safety Analysis Report, Chapter 2, to identify areas of the plant vulnerable to flooding and containing safety equipment. A general site walkdown was conducted, with a specific walkdown of the emergency diesel generator rooms, to ensure that flood protection measures were in accordance with design. Specific attributes verified included sealing of penetrations below the design floodline, adequacy of watertight doors between flood areas, a check that the site embankment remained intact, and verification that no unanalyzed sources of internal flooding were in place.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (ISI)

a. Inspection Scope

The inspectors observed in-process ISI work activities and reviewed selected ISI records. The observations and records were compared to the Technical Specifications (TS) and the applicable Code (ASME Boiler and Pressure Vessel Code, Sections V and XI, 1989 Edition, with no Addenda) to verify compliance.

Portions of the following ISI examinations were observed:

Ultrasonic (UT) Main steam piping welds MS-6A, MS-287, MS-288.

Magnetic Particle (MT) Steam generator A nozzle weld Mk14-3 and Pressurizer support weld Mk 126/128-3.

Qualification and certification records for examiners, equipment and consumables, and nondestructive examination (NDE) procedures for the above ISI examination activities were reviewed. In addition, the licensee's most recent audit of the inservice inspection program was reviewed for effectiveness.

The inspectors observed performance of field snubber functional tests for Reactor Coolant System support RCH-620 and bench machine testing of Reactor Coolant System support, RCH-64, and Main Steam System support, MSH-167. The inspectors observed the licensee's quality control (QC) inspectors witnessing the snubber functional testing. The inspectors reviewed the training, medical, and qualification records for the QC inspectors and machine operators for the snubber functional tests for adequacy.

In addition to the above observations and reviews for the current Unit 3 outage, the inspectors observed activities relative to UT examination of nine control rod drive mechanism (CRDM) nozzles. These UT examinations were performed as the result of leakage identified on CRDM 32 which had been identified during visual examinations of Unit 3 reactor vessel head penetrations (VHPs) in response to NRC Bulletin 2001-01. The inspection included review of contractor UT examination procedures, assessment of contractor NDE personnel training and qualification, and observation and assessment of in-process UT examinations. In addition, licensee's evaluation of UT examinations in the licensee's corrective action program were reviewed. The activities were examined to verify licensee compliance with regulatory requirements and to gather information to help the NRC staff identify possible further regulatory positions and generic communications.

<u>Once Through Steam Generator (OTSG) Inspection</u>

The inspectors reviewed selected inspection records for the eddy current examination of the OTSGs. The records were compared to the Technical Specifications (TS), License Amendments and applicable Industry Established Performance Criteria to verify compliance. Qualification and certification records for examiners, equipment and procedures for the above eddy current examination activities were reviewed.

<u>Control Rod Drive Mechanism (CRDM) Nozzle 32 Weld Repair</u>

The inspectors reviewed the documentation for the inner diameter (ID) temperbead weld repair of CRDM Nozzle 32. The review included the Welding Procedure Specification, the supporting Procedure Qualification Record (PQR) and the weld control record for the repair weld. The procedure qualification and repair welding documentation were inspected for conformance to the ambient temperature temper bead rules of ASME Section III, NB, 1989 Edition No Addenda; Section XI, IWA & IWB, 1989 Edition No Addenda, and Section XI Code Cases N-416-1 and N-638.

b. <u>Findings</u>

No findings of significance were identified.

1R11 <u>Licensed Operator Requalification</u>

.1 <u>Simulator Training</u>

a. <u>Inspection Scope</u>

The resident inspectors observed a requalification examination simulator session to verify that operator performance was consistent with 10 CFR 55 requirements and industry guidelines and that licensee evaluators properly implemented 10 CFR 55.59 requirements. During this session, the inspectors assessed the crew's abilities in making emergency classifications and notifications as part of the conduct of emergency operations.

b. <u>Findings</u>

No findings of significance were identified.

.2 <u>Requalification Program</u>

a. <u>Inspection Scope</u>

The inspectors reviewed the facility operating history and Licensee Event Reports since the last requalification program inspection for indications of operator weaknesses and their inclusion in the feedback process. The inspectors also reviewed the biennial written examinations administered last year and evaluated their effectiveness in providing a basis for assessing operator knowledge of material covered in the requalification training program. Examination quality, licensee effectiveness in integrating industry events, plant and student feedback into the requalification training program, and examination development methodology were evaluated for proper implementation of 10 CFR 55.59 requirements. The inspectors observed three annual dynamic simulator examinations for 17 operators to assess the adequacy of the licensee's evaluation of operator knowledge and abilities. During these observations, the inspectors assessed licensee evaluator effectiveness in pinpointing operator performance deficiencies requiring supplemental training. The inspectors also evaluated and observed 23 Job Performance Measures (JPMs) as part of the walkthrough examination administered by the licensee during this requalification segment to assess evaluator performance.

The inspectors reviewed a sample of licensed operator requalification attendance records, watchstanding records, reactivation records, and twenty five percent of the licensed operator medical records to ensure compliance with 10 CFR 55.59, Requalification and 10 CFR 55.53, Conditions of Licenses.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation

a. Inspection Scope

The inspectors reviewed nuclear condition report (NCR) 48950 concerning the cracking of the control rod drive mechanism to reactor vessel head nozzle weld. The inspectors assessed whether licensee's maintenance rule scoping for the reactor coolant system (RCS) was in accordance with 10 CFR 50.65 and that the specified problem was characterized in the licensee's corrective action program. The inspectors checked the licensee's maintenance rule program (a)(1) classification for the reactor coolant system to assure consistency with licensee compliance procedure CP-153B, Monitoring the Performance of Structures, Systems, and Components Under the Maintenance Rule and for consistency with 10 CFR 50.65 requirements. The inspectors checked the Final Safety Analysis Report; Technical Specifications; and the licensee's Maintenance Rule Scoping Report for consistency with the maintenance rule classification and action plans.

The inspectors reviewed NCR 48844 concerning the normal makeup control valve MUV-31 failing in the closed position. The inspectors assessed whether licensee's maintenance rule scoping for RCS inventory control and purification was in accordance with 10 CFR 50.65 and that the specified problem was characterized in the licensee's corrective action program. The inspectors checked there was no actual loss of function with the bypass valve MUV-30 available in conjunction with valve MUV-51 controlling letdown flow. The inspectors checked the Final Safety Analysis Report; Technical Specifications; and the licensee's Maintenance Rule Scoping Report for consistency with the maintenance rule classifications and action plans.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

a. Inspection Scope

The inspectors reviewed daily maintenance schedules and observed work controls to evaluate risk before maintenance was conducted. The inspectors employed standards for operability of equipment such as those found in Technical Specifications, the Final Safety Analysis Report, licensee procedures, and regulatory information such as NRC Generic Letter 91-18, Revision 1, Information to Licensees Regarding NRC Inspection Manual Section on Resolution of Degraded And Nonconforming Conditions. The inspectors also reviewed maintenance schedules to assure that overall risk was minimized through preservation of safety functions such as decay heat removal capability, reactor coolant system inventory control, electric power availability, reactivity control, and primary containment control. The inspectors assessed whether licensee personnel were managing risk by assuring that key safety functions were preserved and that upon identification of an unplanned situation, the resulting emergent work was evaluated for risk and controlled as described in Technical Specifications, licensee Compliance Procedure CP-253, Power Operations Risk Assessment and Management, and Operations Instruction OI-7, Control of Equipment and System Status. The inspectors verified that risk significant emergent work was documented in the corrective action program. The inspectors evaluated risk controls associated with nuclear condition report 52209 which was written when containment penetration number 430 expansion chamber rupture disk (CARS-3) was found ruptured while an emergency feedwater pump was out of service for scheduled maintenance. The inspector determined whether the licensee took actions specified by Technical Specifications.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the technical adequacy of nuclear condition report NCR 49670 to verify that operability of electrical breaker 3205 on the A train vital 4160 volt bus was consistent with Technical Specifications, the Final Safety Analysis Report, 10CFR Part 50 requirements, and NRC Generic Letter 91-18, Revision 1, Information to Licensees Regarding NRC Inspection Manual Section on Resolution of Degraded And Nonconforming Conditions. The inspectors monitored licensee activities to verify that operability issues were being identified at an appropriate threshold, consistent with 10 CFR 50, Appendix B requirements, and licensee procedure NGGC-200, Corrective Action Program, and that risk was assessed when plant problems were identified.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications

a. Inspection Scope

The inspectors evaluated six modifications in the Initiating Events, Mitigating Systems, and Barrier Integrity cornerstone areas, to verify that the modified systems' designs had not been degraded, and that the modifications had not left the plant in an unsafe condition. The inspectors verified the following inspection attributes were satisfied: energy requirements can be supplied by supporting systems; materials/replacement components are compatible with physical interfaces; replacement components are seismically qualified for application; safety classification of replacement system, structures, and components were consistent with design bases; the appropriateness of modification design assumptions; that post modification testing would establish operability; those failure modes introduced by the modification are bounded by existing analyses; and that appropriate procedures or procedure changes have been initiated.

The major documents reviewed are listed in this report that included corrective action documents, drawings, procedures, testing documents, installation packages, and calculations. The inspectors also reviewed additional information as necessary such as applicable sections of the Final Safety Analysis Report (FSAR), the FSAR update packages, design basis documentation, supporting analyses, technical specifications, and procedures.

In addition, the inspectors reviewed licensee audit and assessment reports to confirm that the licensee was identifying modification issues and initiating actions to resolve concerns.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors evaluated the following post-maintenance testing activities for risk significant systems to assess the following (as applicable): (1) the effect of testing on the plant had been adequately addressed; (2) testing was adequate for the maintenance performed; (3) acceptance criteria were clear and demonstrated operational readiness; (4) test instrumentation was appropriate; (5) tests were performed as written; and (6) equipment was returned to its operational status following testing. The inspectors evaluated the licensee activities against the Technical Specifications, the Final Safety Analysis Report, 10 CFR Part 50 requirements, licensee procedures, and various NRC generic communications.

The specific post-maintenance activities evaluated included:

- Surveillance Procedure SP-102, Control Rod Drop Time Testing, after PT-445, Control Rod Programming Verification; and PM-114, Control Rod Drive Mechanism - Electrical Checks after a rod group transfer relay was replaced.

- Surveillance Procedure SP-354B, Monthly Test of EGDG-1B, including Section 4.6, Maximum Load Testing, following scheduled outage tear-down and rebuild of emergency diesel generator EGDG-1B.

- Surveillance Procedure SP-349C, Emergency Feedwater Pump Number 3 and Valve Surveillance, following planned maintenance on emergency feedwater pump number 3.

b. Findings

No findings of significance were identified.

1R20 Outage Activities

a. Inspection Scope

The inspectors attended daily outage control meetings to verify that the licensee controlled outage risk in accordance with their risk management plans and outage schedule. During periods of heightened risk, such as the reduced inventory condition, the inspectors conducted walkdowns to verify that the licensee maintained redundancy in decay heat removal, inventory control, reactivity control, and electrical power availability. When in reduced inventory, the inspectors checked if the licensee maintained key safety functions as described in NRC Generic Letter 87-12, Loss of Residual Heat Removal While the Reactor Coolant System is Partially Filled, and assured that containment could be established within the licensee's estimated time to reactor water boiling should decay heat removal be inadvertently lost.

The inspectors conducted periodic walkdowns of reactor containment to verify that work controls, personnel safety, radiation controls, and foreign material controls were consistent with licensee procedures and outage plans. During significant maintenance on vital plant equipment, such as the emergency diesel generators, the inspectors assessed whether the reactor was in a stable flooded condition and that adequate electrical redundancy was available for core cooling. The inspectors determined whether electrical safety tagout 01-13-095, for the 4160 volt, A Engineered Safeguards Bus, provided adequate electrical isolation and was consistent with electrical drawing EC-206-011, Electrical One-line Diagram.

During the preparations for return to power operation, the inspectors conducted tours of containment and checked the licensee's completed surveillance SP-324, Containment Inspection, to assure that foreign material controls supported operability of the containment sump. Prior to entry into operational modes 3 and 2, the inspectors reviewed the following surveillance tests to verify that the limiting condition for operation allowed time for these tests had been completed in accordance with Technical Specification 3.0.4.

- SP-130, Engineered Safeguards Monthly Functional Test (Technical Specification 3.3.5.2)
- SP-332, Monthly Steam Line and Feedwater Functional Test (Technical Specification 3.3.13.1)
- SP-344C, Containment Cooling System Fan and Valve Test (Technical Specification 3.6.6.3)
- SP-349B, Emergency Feedwater Pump (EFP-2) and Valve Surveillance (Technical Specification 3.7.5.1)
- SP-102, Control Rod Drop Time Test (Technical Specification 3.1.4.3)

On October 24, with the reactor in Mode 3, following an automatic emergency feedwater actuation, the inspectors responded to the control room to evaluate if the emergency feedwater had actuated as designed and to check that the system operated automatically to control OTSG level. During this time, the inspectors walked down the operating emergency feedwater pump (EFP-3) to check that the engine ran smoothly with no abnormal indications or alarms. When auxiliary steam was returned and main feedwater was returned to service, the inspectors assessed whether the licensee returned the emergency feedwater initiation and control system to the automatic (standby) mode. The inspectors verified that the licensee entered this event into their corrective action system.

On October 25, 2001, the inspectors observed portions of the licensee's restart operations including shutting the main electrical output breakers returning the reactor to power operation, to verify that the evolution was conducted in accordance with plant technical specifications and operating procedures.

b. Findings

No findings of significance were identified.

.2 Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed the licensee's actions to disposition and correct an incident involving a potential loss of 4160 volt bus. The inspectors evaluated the licensee's cause determination and verified associated corrective actions against NRC regulations.

b. Findings

(Green) The inspectors found that the licensee investigation of a potential loss of 4160 bus that involved worker performance issues did not consider worker fatigue in the assessment.

On October 9, 2001, a journeyman electrician, under supervision of a licensee electrician, started work on the wrong (energized) side of the open B ES transformer supply breaker which was providing power to the operating decay heat removal equipment. The licensee determined that the individual's behavior contributed to the event when the wrong equipment was accessed and zero energy checks were not performed. The individual contacted energized 4160 volt equipment, received an electrical shock and burns to the arm and forearm, and was hospitalized. The enforcement associated with the licensee identified failure to properly control work activities is discussed in Section 4OA7.

The inspectors reviewed the licensee investigation of event (Nuclear Condition Report (NCR) 49526) and determined that although having identified worker performance issues, the potential contribution of worker fatigue was not considered in the licensee evaluation of event. The inspector determined that the job supervisor had worked five 13 hour nights, one 12 ½ hour night and one 9 hour night in the seven days prior to the event and had received a waiver from the Technical Specification overtime restrictions. This was not assessed by the licensee's investigation. This finding is more than minor because the event was viewed as a precursor to a significant event (loss of decay heat removal) and affected the initiating event cornerstone. The finding is considered to be of very low safety significance (Green) because no actual loss of safety equipment occurred. The inspectors did not identify that this finding involved a violation of NRC requirements.

The inspectors determined that neither the worker nor the supervisor had been For-cause tested in the Licensee Fitness for Duty Program following the event. This issue is being treated as Unresolved Item (URI) 50-302/01-04-01, Fitness for Duty Testing for -Cause, pending further NRC review of 10 CFR Part 26, Section 24(a)(3) requirements. The issue is in the licensee corrective action program as NCR 52293.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors observed surveillance testing (SPs) or reviewed test data for risk-significant systems or components, to assess compliance with Technical Specifications, 10 CFR Part 50, Appendix B, and licensee procedure requirements. The testing was also evaluated for consistency with the Final Safety Analysis Report, NRC Generic Letter 89-04, Guidance on Developing Acceptable Inservice Testing Programs, and NUREG-1482, Guidelines for Inservice Testing at Nuclear Power Plants. The inspectors checked if the testing demonstrated that the systems were ready to perform their intended safety functions. During the inspections, consistent with 10 CFR Part 50, Appendix B, Criterion XVI, and licensee procedure CAP-NGGC-200, Corrective Action Program, the inspectors verified that licensee personnel were documenting surveillance problems in the corrective action program.

Inservice test (IST) activities were reviewed to ensure testing methods, acceptance criteria, and required corrective actions were in accordance with the ASME Code, Section XI, and Florida Power Corporation ASME Section XI, Ten Year Inservice Testing Program, dated May 4, 1998. The specific surveillance activities assessed included:

SP-353, Control Room Emergency Ventilation and RM-A5 Monthly Test
SP-524, Battery Performance Discharge Test
SP-341, Monthly Containment Isolation Valve Operability Check
SP-349C, Emergency Feedwater Pump Number 3 and Valve Surveillance (IST)
SP-630, Makeup Pump/High Pressure Injection Check Valves Full Flow Test

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation

a. Inspection Scope

The inspectors observed conduct of a November 16, 2001, operator requalification examination in the plant specific simulator. The inspectors assessed whether the crew correctly classified the event and made notifications of a Site-Area-Emergency following a simulated steam generator tube rupture and offsite release as specified by the pre-scripted scenario and in accordance with the Crystal River Radiological Emergency Response Plan, Section 8.0, Emergency Classification System, and 10 CFR Part 50.72 and 10 CFR Part 50, Appendix E. The need for protective action recommendations was checked using licensee emergency response procedures. The inspectors attended the post-scenario critique to check that the licensee evaluated the crew in accordance with the Radiological Emergency Response Plan. The inspectors also assessed whether conduct of emergency operations and crew communications were in accordance with licensee procedures.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY
Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas

a. Inspection Scope

During the week of October 01, 2001, administrative and engineering controls were evaluated for high radiation and locked-high radiation area Refueling Outage (RFO) 12 activities conducted in accordance with the following Radiation Work Permits (RWPs):

- RWP No. 2040, Shielding Installation/Removal, Reactor Building
- RWP No. 2047, Scaffolding Installation/Removal, Reactor Building
- RWP No. 2056, Reactor Head Work in Cavity and Head Movement
- RWP No. 2057, Reactor Head Work on Stand Activities
- RWP No. 2060, Index Fixture and Plenum Movement
- RWP No. 2062, Once-Through Steam Generator (OTSG) Hand Hole & Manway Removal/Installation
- RWP No. 2062, OTSG Nozzle Dam Removal/Installation
- RWP No. 2062, OTSG Eddy Current/Tube Repair/Plugging/Removal

Evaluations were conducted through attendance at pre-job briefings, review of current status of planned tasks, assessment of personnel exposures, and observations of work-in-progress and Health Physics (HP) technician job coverage. Conduct of selected radiation and contamination surveys was observed and results discussed. Electronic alarming dosimetry (EAD) setpoints were assessed for selected tasks. Personnel EAD exposure results, contamination event assessments, and internal exposure evaluations were reviewed and discussed with licensee representatives. During tours and observation of reactor building and auxiliary building RFO 12 work activities, the inspectors evaluated administrative and engineering controls for access to high radiation, locked-high radiation, and very high radiation areas. Licensee nuclear condition report 48511, documented on September 25, 2001, regarding a worker lacking dosimetry who entered a reactor building radiation area, was reviewed and evaluated in detail.

Licensee activities were reviewed against Final Safety Analysis Report (FSAR), Technical Specification (TS), and 10 CFR Part 20 requirements.

a. Findings

No findings of significance were identified.

2OS2 "As Low As Reasonably Achievable" Program Planning and Controls

a. Inspection Scope

During the week of October 01, 2001, "As Low As Reasonably Achievable" (ALARA) program implementation for ongoing RFO 12 outage activities were evaluated. The inspectors discussed dose rate and cumulative dose expenditure data trends associated with selected systems, equipment and tasks relative to past refueling outages. Recent revisions to ALARA Work Plan were reviewed, and implementation of selected dose reduction initiatives were observed and their effectiveness evaluated. General dose reduction initiatives reviewed and evaluated included shutdown chemistry and cleanup, worker dose tracking and reporting, system flushes, temporary shielding, and remote worker monitoring capabilities. Knowledge of ALARA program guidance and staff proficiency in program implementation were appraised from observation of selected work activities, comparison of estimated and current dose expenditure data for selected tasks, and discussions of selected outage tasks with responsible supervisors and managers. Implementation and effectiveness of detailed ALARA initiatives and planning were evaluated for the following RFO 12 activities:

- Steam Generator Manways and Handholds
- Steam Generator Nozzle Dam Installation and Removal
- Reactor Head Removal, Maintenance, Replacement
- Reactor Head Nozzle Inspection, Cleaning, and Repair
- Reactor Building Scaffolding and Insulation
- Health Physics Refuel 12 Outage Activities

Program implementation and effectiveness were reviewed against the facility's ALARA work plans, FSAR, 10 CFR Part 20 requirements, and technical specifications.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

4OA1 Performance Indicator Verification

a. Inspection Scope

The inspectors checked the accuracy of the performance indicators for safety system functional failures, unplanned power changes, and high pressure injection system unavailability. Performance indicator data submitted in October 2001, were compared for consistency to data obtained through the review of control room logs, monthly operating reports, and equipment out-of-service records from October 2000 through September 2001. The inspectors verified that relevant issues related to the collection of performance indicator data had been entered into the licensee corrective action program and corrected.

a. Findings

No findings of significance were identified.

4OA2 Problem Identification and Resolution

The inspectors found that the licensee investigation of a potential loss of 4160 volt bus that involved worker performance issues did not consider worker fatigue in the assessment (Green). (Section 1R20)

4OA3 Event Followup

(Closed) Licensee Event Report (LER) 50-302/01-002: Main Steam Safety Valve Setpoints Outside Required Tolerance Longer Than Allowed by Technical Specifications

On September 26, 2001, and on September 28, 1999, the licensee identified that two main steam safety valves were outside the maximum lift setpoint tolerance specified in Improved Technical Specification Table 3.7.1-1, Criterion D. Florida Power Corporation stated in the LER that the valves had been out of specification during reactor operation and the actions of Technical Specification 3.7.1 to reduce thermal power output and reset the nuclear overpower trip setpoint had not been taken. In the LER, the licensee found that although the safety valves setpoints had drifted, the total available relieving capacity exceeded the required relieving capacity for overpressure protection. The condition did not result in a reduction in safety and was in the licensee corrective action system as NCR 48648. This finding is more than minor because it had a credible impact on safety because if additional safety valves were found outside their allowed tolerance, then steam generator integrity could not be assured for possible reactor accidents. This finding affects the Barrier Integrity Cornerstone and was considered to have very low safety significance (green) because the likelihood of an accident leading to core damage was not affected, the probability of steam generator failure was negligible, and the steam generators remained intact. The licensee also identified in the LER, that the 1999 condition was not reported within the time requirements of 10 CFR 50.73 (NCR 51139). The licensee identified non-cited violations are discussed in Section 4OA7. This LER is closed.

4OA5 (Closed) NRC Temporary Instruction (TI) 2515/145, “Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles (NRC Bulletin 2001-0.1)

a. Inspection Scope

The inspectors reviewed the Crystal River 3 visual inspection program for reactor vessel head penetrations as discussed in the licensee’s response to NRC Bulletin 2001-01. The inspection guidelines were provided in TI 2515/145.

b. Findings

1) Verification that visual examination was performed by qualified and knowledgeable personnel:

The inspectors verified the ASME VT-2 qualifications for the personnel responsible for performance of the visual examinations at Crystal River Unit 3. In addition, the inspectors verified that examination personnel had received specialized industry-developed training on the visual examination methods for leakage of reactor head penetrations and on the site specific procedures to be used for the examinations. The inspectors interviewed the examination personnel and noted that they were knowledgeable of the specialized qualification criteria. The inspector verified that all examination personnel were certified as Level II or III, VT-2.

2) Verification that visual examination was performed in accordance with approved and adequate procedures:

Before the examination was conducted, the inspectors verified the adequacy of Florida Power Special Process Specification, SPS VT-N14, Visual Examination of System Pressure Testing ASME Code Section XI; and Florida Power Specification SPS VA-N11, Visual Acceptance of System Pressure Testing ASME Section XI for conduct of the VHP visual examination. The inspectors observed that the examination was done using these procedures under Work Request 368781. The inspectors verified by direct observation and in discussions with examination personnel that the approved acceptance criteria and/or critical parameters for VHP leakage were applied in accordance with the procedures.

3) Verification that the licensee was able to identify, disposition, and resolve deficiencies:

The inspectors verified that the licensee’s inspection plan provided nozzle indexing and drawings with adequate guidance to ensure that the visual examinations included 100% circumferential coverage of each VHP. The inspectors verified that the examination result for each penetration was individually documented. The examination procedure provided acceptance criteria for the VT-2 examination with specific follow-up actions for the detection of boric acid residues or identified leakage. The procedure required that questionable control rod drive mechanism penetration leakage be identified as a leaking nozzle. One leaking nozzle was identified in the examination (Penetration 32). This nozzle penetration exhibited bright white, popcorn like crystals of boron that were extruded from the nozzle annulus around the penetration. No other nozzle penetration had similar indications.

4) Verification that the licensee was capable of identifying the Primary Water Stress Corrosion Cracking (PWSCC) phenomenon described in the bulletin:

The inspectors visually observed the vessel head prior to the licensee's examination; observed the licensee conduct the examination; discussed the examination with the licensee examiners prior to, during, and following the examination; reviewed the documentation and verified 100% circumferential coverage of each VHP; and verified the qualification of the licensee examination personnel. The inspectors concluded that the licensee conducted an effective visual inspection to identify potential leakage resulting from PWSCC cracking of VHP nozzles.

5) Evaluate condition of the reactor vessel head (debris, insulation, dirt, boron from other sources, physical layout, viewing obstructions):

The inspectors observed no significant examples of insulation, leakage sources, debris, dirt, or other physical impediments that prevented a thorough visual examination. The licensee was able to adequately view each of the 69 control rod drive mechanism nozzles during the visual examinations.

6) Evaluate ability for small boron deposits, as described in the bulletin, to be identified and characterized:

The inspectors observed that the reactor head was generally free of any deposits that would have hindered the visual examination. Some loose corrosion products possibly mixed with dark-colored boric acid flakes were observed and readily removed by the licensee to allow complete viewing during the examinations. In three instances (Penetrations 40, 63, and 69) the licensee examiners noted a thin layer of boron about the VHP annulus area. In each case, these observations were noted in the examination record and each penetration received additional inspection by the licensee that included scraping the deposits to ensure that the source was from above and not from the annulus, below. Two of these nozzle penetrations (40, 63) were subsequently examined by ultrasonic testing and no cracks were found. With exception of nozzle 32, no significant examples of boron were identified during the inspection. No localized corrosion was noted in any area.

7) Determine extent of material deficiencies (associated with the concerns identified in the bulletin) which were identified that required repair:

The inspectors observed that VHP Nozzle 32 exhibited popcorn-like, extruding boron deposits similar to those observed at Oconee and Arkansas Nuclear Unit 1. The licensee confirmed the existence of a flaw in Penetration 32 using ultrasonic testing and in accordance with their response to the NRC bulletin. The licensee examined eight other VHPs using ultrasonic testing and no other flaws were identified.

8) Determine any significant items that could impede effective examinations and/or ALARA issues encountered

The inspectors noted no ALARA issues or examples of significant items that could impede the visual examination process.

4OA6 Meetings

Exit Meeting Summary

The resident inspectors presented the inspection results to Mr. D. Young and other members of licensee management at the conclusion of the inspection on January 14, 2002. The inspectors asked the licensee whether any of the material examined during the inspection should be considered proprietary. The licensee did not identify any proprietary information.

4OA7 Licensee Identified Violations

The following findings of very low significance were identified by the licensee and are violations of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600 for being dispositioned as NCVs.

If you deny any of the non-cited violations, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001, with copies to the Regional Administrator, Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Crystal River 3 facility.

NCV Tracking Number	Requirement Licensee Failed To Meet
NCV 50-302/01-04-02	10 CFR Part 50, Appendix B, Criterion V, states that activities affecting quality shall be prescribed by documented procedures of a type appropriate to the circumstances and shall be accomplished in accordance with these procedures. Contrary to this requirement, on October 9, 2001, during maintenance on 4160 volt engineered safeguards (ES) bus 3A using Work Request 365187: 1) Although the work request stated “Check Bus for Voltage before starting work”, dead bus checks were not done; 2) Although licensee Administrative Instruction AI-610, Electrical Safety, required a maintenance risk assessment be performed on all work on energized equipment with the work assessed as medium or high risk, the work request had not been risk assessed and was classified low risk; and 3) Although licensee Administrative Instruction AI-504, Guidelines for Cold Shutdown and Refueling, stated “Power supplies (for operating safety equipment shall be) controlled by physical barriers with signs” an energized power supply for the operating decay heat removal equipment accessed by a worker was not controlled by a physical barrier with a sign. This was identified in the licensee’s corrective action program as CR-42306 (Green).

NCV 50-302/01-04-03	Technical Specification 3.7.1.A requires that all main steam line code safety valves shall be operable with lift settings as specified in Table 3.7.1-1. If one or more main steam line code safety valves are inoperable, actions must be taken in accordance with Action A of Technical Specification 3.7.1. Contrary to the above, as of September 26, 2001 and September 29, 1999, main steam line safety valves did not have lift settings in accordance with Table 3.7.1-1, and the requirements of Action A of Technical Specification 3.7.1 were not met. This condition was identified by the licensee and documented in Nuclear Condition Report NCR 48648. This condition was reported in LER 50-332/01-002 (Green).
NCV 50-302/01-04-04	10 CFR 50.73 (a)(2)(B), requires that any condition prohibited by plant technical specifications shall be reported by the licensee in a Licensee Event Report within 60 days after discovery of the event. Contrary to the above, the licensee determined that two main steam safety valves had setpoints outside of the Technical Specification Table 3.7.1-1 required tolerance, the actions of the technical specification were not taken, and the condition was not reported within 60 days after discovery (September 1999). This condition was identified by the licensee and documented in Nuclear Condition Report NCR 51139 (No Color).

PARTIAL LIST OF PERSONS CONTACTED

Florida Power Company

M. Annacone, Manager, Operations
S. Bernhoft, Manager, Regulatory Affairs
G. Chick, Manager, Outages and Scheduling
R. Davis, Manager, Training
C. Gurganus, Manager, Maintenance
J. Holden, Director Site Operations
S. Johnson, Supervisor, Self-Evaluation
F. Marcussen, Superintendent, Security
S. Powell, Supervisor, Licensing
D. Roderick, Plant General Manager
J. Stephenson, Supervisor, Emergency Preparedness
J. Terry, Manager, Engineering
R. Warden, Manager, Nuclear Assessment
D. Young, Vice President, Crystal River Nuclear Plant

NRC
J. Monninger, Acting Branch Chief, NRC Region II
J. Wallo, Physical Security Inspector, NRC Region II

ITEMS OPENED AND CLOSED

50-302/01-04-02	NCV	Failure to Follow Procedures During Electrical Maintenance on the Engineered Safeguards Bus (Section 40A7)
50-302/01-04-03	NCV	Main Steam Safety Valve Setpoints Outside Required Tolerance Longer than Allowed by Technical Specifications (Section 4OA7)
50-302/01-04-04	NCV	Failure to Report a Condition Prohibited by Technical Specifications (Section 4OA7)

ITEMS CLOSED

50-302/01-002	LER	Main Steam Safety Valve Setpoints Outside Required Tolerance Longer than Allowed by Technical Specifications (Section 4OA3)
TI 2515/145		Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles (NRC Bulletin 2001-0.1) (Section 4OA5)

ITEM OPENED

50-302/01-04-01	URI	Applicability of 10 CFR Part 26, Section 24(a)(3) requirements (Section 1R20)

List of Documents Reviewed for 1R08

Procedures

Crystal River Steam Generator Integrity, Rev. 1, (Effective Dates 8/14/1998 - 8/13/2008) June 27, 2001
Framatome 51-5005589-01, Qualified Eddy Current Examination Techniques for Crystal River Unit #3.
Work Package No. 370040, CRDM Nozzle #32 Weld Repair Documentation
Framatome Welding Procedure Specification (WPS) 55-WP3/F43TBSCa3-01, 9/12/01
Framatome Procedure Qualification Record (PQR) PQ7164-00, 9/4/01
Calculation Summary Sheet (CSS) 32-5014980-00, CR-3 Tube End Cracking ARC Leakage Calculation - 12 RFO
Crystal River Unit 3 CRDM Nozzle Ultrasonic Examination Results
Framatome Nondestructive Examination Procedure, 54-ISI-100-06, Remote Ultrasonic Examination of CRDM Nozzles
Special Process Specification, SPS - UT- N16, Ultrasonic Examination of Ferrite Steel Piping Welds
Special Process Specification, SPS - MT- N01, Dry Visible Magnetic Particle Examination
Plant operating Manual MP-175, Power Piping Pipe Snubber Removal and Installation
Surveillance Procedure SP-201, Hydraulic Snubber Visual Inspection
Plant Operating Manual SP-208, Visual Examination of Component Supports
Test Procedure 46048-10, Procedure for Functional Testing of Hydraulic Snubbers Using the Wyle STM Model 100 Test Machine for Crystal River 3
Plant Operating Manual MP-400, Barker/Diacon 130 Kip Bench Tester Model S-4000 Upgrades

Vendor Exam Evaluation Reports (VEs)

VE-01-019, Acceptance of fabrication type UT indications on CRDM 8 nozzle
VE-01-020, Acceptance of fabrication type UT indications on CRDM 21 nozzle
VE-01-021, Five crack like indications identified during UT exams on CRDM 32 nozzle, cracks required repair
VE-01-022, No UT indications on CRDM 40 nozzle
VE-01-023, No UT indications on CRDM 52 nozzle
VE-01-024, Acceptance of fabrication type UT indications on CRDM 54 nozzle
VE-01-025, Acceptance of fabrication type UT indications on CRDM 58 nozzle
VE-01-026, Acceptance of fabrication type UT indications on CRDM 63 nozzle
VE-01-027, No UT indications on CRDM 64 nozzle

Other Documents

Florida Power Corporation Nuclear Quality Assessment Report, 00-02, Inservice Inspection

List of Documents Reviewed for 1R02 and 1R17

Note: * indicates modifications examined for 1R17

10CFR50.59 SCREENS

DOCUMENT	50.59 #	DESCRIPTION
MARs		
*99-10-02-01	99-0415	Constant Level Oilers on DHP1A&1B
00-02-03-01	N/A NEP-210A	Replace DJP-3 and DJP-4
*00-05-04-01	00-0204	Pressurizer Heater Group 3 to 9 SWAP
00-06-06-01	00-0268	AEH Control System Upgrade
*00-09-03-01	N/A NEP-210A	Replace BAST Level Transmitters
01-03-02-01	01-0049	Changes to WDV Position Interlock Logic for MU System Feed Permit
01-10-02-01	N/A NEP-210A	DFP-3A Motor Overload Heater Replacement (On Hold)
Calculations		
F-98-0013, Rev 3	01-0188	Revising PTLR down to 100 degrees F for RCP Operation (3F0901-06)
Procedures		
OP-103B, Rev 31 SP-300, Rev 172	01-0046	EEM-01-006, RW/SW Heat Exchanger Degradation Compensatory Actions
OP-103B, Rev 31 SP-300, Rev 172	01-0095	Restoration of Ultimate Heat Sink (UHS) Limits in Procedures
EOP-4, Rev 7	01-0228	Transition from EOP-4 to EOP-3 on a Loss of SCM
Corrective Actions Documents		
PC00-0831	01-0054	OCR 01-0002, EFP-3 Air Filter Differential Pressure Monitoring (SP-349C)

10CFR50.59 EVALUATIONS

DOCUMENT	50.59 #	DESCRIPTION
MARs		
*86-09-22-14	99-0071	Replacement of Radiation Monitors RM-G26 and RM-G27 with upgraded detectors (PCs 97-4661, 97-4662 and 98-5657 relate) Leaving the Containment Mini Purge Open
00-02-04-02	00-0315	ES MCC 3AB Refurbishment
*00-08-05-01	01-0034	CRDM Drive Replacement
00-07-05-01	01-0016	Mark B10 Fuel Assemblies
*01-09-01-01	01-0317	RCP Seal Cooler Plugs
Calculations		
M-90-0021, Rev 11	99-0382	Building Spray and Decay Heat NPSH
M-94-0040, Rev 2	1-0177	REA 01-0103, MUV-541 Seat Leakage requirement
Technical Specifications		
Bases Change B99-28	99-0433	ITS, EDG Overspeed Trip Point Acceptance Criteria Change for MP-499, Rev 19
Corrective Actions		
PC97-4355	99-0474	LAR 222 (3F1297-19), CREVS and Ventilation Filter Testing Program (Control Room Habitability) [licensee actions pending]
PC00-2487	00-0289	Condition Resolution, Revision of EOPs for 1 Minute RCP Trip [licensee action pending]

Other Supporting Documentation Reviewed

Calculations

EEM-01-006, Revision 0, Evaluation of Bypass Flow in SWHE-1C and SWHE-1D

M-01-0007, Rev 1, Seal Operability During Loss of Seal Injection Transient with a Blocked Reactor Coolant Pump Heat Exchanger

Corrective Actions Documents

PC 3-C01-0266, RWP-1 Discharge Pressure Increased when a Clean SWHE was Placed in Service and a Dirty SWHE was Removed

NCR 00041425, RWP-1 Discharge Pressure Increased when a Clean SWHE was Placed in Service and a Dirty SWHE was Removed

PC 3-C97-8080, Control Complex Chiller Pre-Rotation Vanes

50.59 evaluation 98-208 [LER 98-11, closed], Control Complex Chillers Operated Outside Design Basis

50.59 evaluation 00-0160 CREVS, Control Room Habitability with Fuel Handling Accident [licensee actions pending]

NCR 00049455, Incorrect Overload Heater Size for DFP-3A Motor

Drawings

Flow Diagram, FD-302-661, Make-up and Purification

Procedures

NEP-210A, Rev 7, Enhanced Modification Approval Records

PM-275, Rev 11, General Preventive Maintenance Work

PM-133, Rev 52, Equipment Lubrication and General Inspection

AP-770, Rev 31, Emergency Diesel Generator Actuation

Test Reports

SP-102, Control Rod Drop Time Testing

Work Orders

NU 0362862, Install 3 new, Larger Oiler Bubblers and Support Per MAR 99-10-02-01 and Attached FEWP Instructions [DHP-1A]

NU 0362861, Install 3 New, Larger Oiler Bubblers and Support Per MAR 99-10-02-01 and Attached FEWP Instructions [DHP-1B]

Regulatory Documents

RIN 3150-AA88, 10 CFR Parts 21 and 60, effective dates October 29, 1991

FSAR Change Request 1999-0040 (Typical)