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

Subject: Brunswick Steam Electric Plant, Unit Nos. 1 and 2
Docket Nos. 50-325 and 50-324/License Nos. DPR-71 and DPR-62
Report of 10 CFR 50.59 Evaluations and Commitment Changes

Ladies and Gentlemen:

In accordance with 10 CFR 50.59(d)(2) and 10 CFR 50.4, Carolina Power & Light Company, now doing business as Progress Energy Carolinas, Inc., is providing a report summarizing the changes, tests, and experiments made during the period from March 31, 2002, through March 31, 2004. This report is provided in Enclosure 1. In addition, a summary of commitment changes for the same period, made in accordance with NEI 95-07 Guidelines, are provided in Enclosure 2.

Please refer any questions regarding this submittal to Mr. Leonard R. Beller, Supervisor - Licensing/Regulatory Programs, at (910) 457-2073.

Sincerely,

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1. Changes, Tests, and Experiments
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Changes, Tests, and Experiments

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Changes, Tests, and Experiments

TITLE: Boral Coupon Removal

BRIEF DESCRIPTION:

Boral sample testing to monitor the high density reactor fuel storage system module poison lifetime is being eliminated and associated sample stations removed based on testing results from 1989 and 1995.

SUMMARY OF THE EVALUATION:

Previous testing confirms that the integrity and neutron absorbing capability of the boral remains unaffected by the fuel pool environment. The testing to date demonstrates that no degradation has occurred and the neutron absorbing capability has been calculated to be well beyond the life of the plant.

PLANT REFERENCES:

Updated Final Safety Analysis Report (UFSAR) Change Package 00FSAR-059, Regulatory Affairs Identification Numbering System (RAINS) 00-1719, 0PT-90.11 Revision 9

Changes, Tests, and Experiments

TITLE: Reactor Building Sample Station Chiller Replacement

BRIEF DESCRIPTION:

The Units 1 and 2 Reactor Building Sample Station chillers were replaced and relocated.

SUMMARY OF THE EVALUATION:

The sample station cooling systems control the temperature of the on-line sample streams. The equipment involved is non-safety related and not required for the safe startup, operation, shutdown, or post accident situations. None of the equipment will impair the operability of accident initiating or accident mitigating systems or equipment. This modification will not reduce the margin of safety designed into the plant.

PLANT REFERENCES:

Engineering Service Requests 0000184 and 0000332, RAINS 01-0184 and 01-0047

Changes, Tests, and Experiments

TITLE: Service Water Valve Freeze Seal Installation

BRIEF DESCRIPTION:

Freeze seals were installed on lines 1-SW-490-6-046 and 2-SW-490-06 to facilitate replacement of the Emergency Diesel Generator (EDG) #1 jacket water cooler isolation valves 1-SW-V679 and 2-SW-V679. These valves had been exhibiting excessive seat leakage.

SUMMARY OF THE EVALUATION:

Installation of the freeze seals has no impact on the operability of the remaining EDGs. The freeze seals serve as an isolation pressure boundary from an operational perspective. The freeze seals were installed in a vertical section of piping, and thus, the probability of a freeze plug related event causing damage to the remaining EDGs is considered negligible given that the ice plug will break loose and follow the direction of flow from the main supply header. The use of the freeze seals has been seismically analyzed and does not present a seismic concern. Controls were established to prevent pipe overpressurization during the ice plug formation. The loss of Service Water (SW) cooling has been evaluated in the UFSAR, and the use of a freeze seal as an isolation boundary poses no increased probability of an accident. Compensatory measures were established, including shutting SW header isolation valves and installing blind flanges. The valves requiring replacement were replaced in series; therefore, any potential failure scenario would only include the loss of SW from one six-inch line. As such, neither the proposed activity nor potential failure scenarios would result in the increase in frequency of occurrence of an accident previously evaluated.

PLANT REFERENCES: Engineering Change (EC) 49151, RAINS 02-0762

Changes, Tests, and Experiments

TITLE: Alternate Source Term Implementation

BRIEF DESCRIPTION:

As a part of the Extended Power Uprate Project, new radiological dose consequence calculations in accordance with the alternate source term (AST) methodology as defined in NRC Regulatory Guide 1.183 were performed . ECs 47810 and 46907 implement the configuration control, design basis and licensing basis changes needed to implement the AST dose calculations for Units 1 and 2.

SUMMARY OF THE EVALUATION:

The new AST radiological consequence analyses for loss of coolant, main steam line break, and control rod drop accidents utilize the methodology of Regulatory Guide 1.183. This regulatory guide provides methods acceptable to the NRC for the post-accident dose consequence analyses and provides the basis for changing to the AST dose analysis.

PLANT REFERENCES: EC 47810 and 46907, RAINS 02-0296 and 02-1076

Changes, Tests, and Experiments

TITLE: Unit 1 Extended Power Uprate

BRIEF DESCRIPTION:

EC 46861 evaluates the main steam line (MSL) high flow trip function, revises the simulated thermal power flow biased trip and Rod Block functions for single and two loop reactor recirculation pump operation, and establishes a new licensing basis for the primary containment response to pipe breaks and testing requirements for validation of successful Extended Power Uprate (EPU) implementation.

SUMMARY OF THE EVALUATION:

Detailed evaluations of the reactor, engineered safety features, power conversion, emergency power, support systems, environmental issues, design basis accidents, and previous licensing evaluations were performed. The results of the evaluations determined that EPU implementation does not create a significant challenge to any safety system, nor result in significant radiological environmental effects or non-radiological environmental effects.

PLANT REFERENCES: EC 46861, RAINS 02-0371

Changes, Tests, and Experiments

TITLE: Shutdown Methodology for Alternate Safe Shutdown Procedures

BRIEF DESCRIPTION:

The shutdown methodology for Units 1 and 2 alternate safe shutdown procedures is being changed to eliminate reliance on Thermo-Lag. Alternate shutdown cooling will be provided using automatic depressurization and low pressure injection in certain scenarios.

SUMMARY OF THE EVALUATION:

Analysis of the proposed changes demonstrates that the NRC-accepted alternate shutdown cooling methodologies do not adversely impact the ability to achieve and maintain safe shutdown. The new shutdown methodologies use existing equipment. This equipment will provide mitigation functions consistent with the existing description in the UFSAR. No physical changes to any system, structure, or component are being implemented. Normal plant operations are not being affected; the only affect on plant operations is in response to a fire.

PLANT REFERENCES: EC 48624 and 48625, RAINS 02-1281

Changes, Tests, and Experiments

TITLE: Standby Liquid Control System Squib Valve Shelf Service Life

BRIEF DESCRIPTION:

This change extends the shelf service life of the standby liquid control (SLC) system injection valves from five to 5.5 years.

SUMMARY OF THE EVALUATION:

The six month extension of shelf service life for the SLC squib valves will not cause any significant change in valve reliability.

PLANT REFERENCES: EC 50522, RAINS 02-1552

Changes, Tests, and Experiments

TITLE: Supplemental Spent Fuel Pool Cooling Pump Motor Breakers

BRIEF DESCRIPTION:

The thermal magnetic breakers used to supply electrical power to the primary supplemental spent fuel pool cooling (SSFPC) pump motors are being replaced with magnetic only breakers to eliminate a spurious tripping problem. Cable protection for the feed to the motor skid is being reduced.

SUMMARY OF THE EVALUATION:

Fault protection is provided by the replacement breaker. Motor overload protection will be provided by the local overload relay, fuses, and thermal-magnetic breaker. A cable overload is virtually impossible, but if it occurred, it would deteriorate into a short circuit rapidly and then be cleared by the new breaker. The worst case failure would be the loss of a safety-related motor control center which is bounded by existing UFSAR analyses. Administrative controls will be established to ensure that this change is only in place during cold shutdown and refueling so that reactor building heat load is not adversely affected.

PLANT REFERENCES: EC 51816, RAINS 03-0354

Changes, Tests, and Experiments

TITLE: Emergency Diesel Generator Temperature Switch Disabled

BRIEF DESCRIPTION:

Defective EDG #4 stator temperature switch 2-DG4-TS-6551-4, which has been determined to be obsolete, is being disabled until a replacement switch can be obtained and installed. Electrical power will be disconnected from the switch and temporary controls established to insure that adequate stator temperature monitoring is maintained.

SUMMARY OF THE EVALUATION:

The stator temperature switch only provides an alarm function and, therefore, will not impact the ability of the EDG to start, load, and perform its intended safety function, nor will it impact the ability of the other EDGs to perform their function. The alarm provides advance warning of an imminent failure of the stator and may limit damage to the stator if the EDG is shutdown soon enough. Since the EDG is provided with automatic features for protection against overloads and short circuits, which are typical causes of stator overheating, and because overheating is a time based condition, periodic local observation of the stator temperature indicator is considered acceptable for overall condition monitoring during normal operation. During an accident scenario, when some of the EDG trips are bypassed, the EDG may be operated to failure whether operators are aware of the stator high temperature condition or not. The loss of this alarm capability for the stator high temperature condition does not increase the likelihood of EDG failure.

PLANT REFERENCES: EC 53185, RAINS 03-0903

Changes, Tests, and Experiments

TITLE: Reactor Pressure Vessel Skirt Manhole Cover Evaluation

BRIEF DESCRIPTION:

This EC provides an evaluation of the acceptability of leaving the installed sheet metal covers for reactor pressure vessel (RPV) skirt manhole covers for long-term operation of the plant and provides the required updates to applicable controlled documents.

SUMMARY OF THE EVALUATION:

The sheet metal manhole covers are embedded in the RPV skirt insulation assembly which is seismically mounted. Possible missile affects were evaluated and determined to not cause credible damage to safety related components. Assumptions for postulated accident scenarios previously evaluated in the UFSAR regarding the availability and performance of Emergency Core Cooling System (ECCS) strainers to mitigate an accident involving a pipe break in the drywell will be unchanged by leaving the sheet metal covers. Other assumptions related to radiological releases, including but not limited to reactor power, days of operation, and potential pathways, are not affected by the covers. The covers are not radiological barriers and the assumptions in previously evaluated accidents regarding the availability and performance of ECCS suction strainers to mitigate an accident will be unchanged.

PLANT REFERENCES: EC 50216, RAINS 02-1522

Changes, Tests, and Experiments

TITLE: Unit 1 Turbine Building Ventilation Once-Through Line-up

BRIEF DESCRIPTION:

This EC evaluates operating the Unit 1 turbine building ventilation system in a once-through line-up, with radiation monitoring for the effluent, on a temporary basis during the B115R1 refueling outage.

SUMMARY OF THE EVALUATION:

The turbine building ventilation system is independent of reactor systems and the reactor coolant boundaries. The operation of the turbine building ventilation system as a once-through design will not adversely affect the reactor coolant piping boundary. The turbine building ventilation system does not affect the reliability of the steam piping in the turbine building. Evaluation of radiological effects under AST considerations determined no significant impact to control room or public dose. While not a malfunction of a system important for safety during an accident, the temporary alignment of the turbine building ventilation system is considered an abnormal line-up in that the turbine building effluent is not recirculated and filtered prior to release. The release of the effluent will be monitored by temporary monitoring equipment that will be installed in the exhaust ductwork at the point of release to assure the dose rates at the site boundary and beyond are maintained below regulatory limits.

PLANT REFERENCES: EC 49493, RAINS 03-0016

Changes, Tests, and Experiments

TITLE: Unit 2 Power Range Neutron Monitoring System Replacement

BRIEF DESCRIPTION:

This modification consists of replacing the existing Power Range Neutron Monitoring (PRNM) System with General Electric's Nuclear Measurement Analysis and Control (NUMAC) PRNM System. This system includes the Average Power Range Monitor system, the Rod Block Monitor system, the Low Power Range Monitor system excluding the detectors and signal cables, and the Enhanced Option 1-A (E1A) function. The Option III stability solution is integrated into the PRNM system electronics as an Oscillation Power Range Monitor upscale trip function, which provides automatic detection and suppression of reactor instabilities instead of the existing manual operational region restrictions and Period Based Detection System (PBDS) installed in E1A.

SUMMARY OF THE EVALUATION:

The new equipment has been specifically designed to assure that it fully meets the existing requirements and has been specifically designed to have the same, or more conservative, "fail safe" failure modes as the current system. The replacement equipment is fully qualified to operate in its installed location, and will not affect other equipment. The Option III stability solution provides automatic detection and suppression of reactor instabilities instead of the existing operational region restrictions and PBDS installed in E1A. The Nuclear Regulatory Commission has determined that E1A and Option III Stability Solution are both acceptable long-term solutions for implementation in any type BWR and that both methodologies provide protection systems to prevent violation of the critical power ratio safety limits.

PLANT REFERENCES: EC 46730, RAINS 02-1058

Changes, Tests, and Experiments

TITLE: Primary Containment Isolation Valve Surveillance

BRIEF DESCRIPTION:

This change replaces Units 1 and 2 Technical Requirements Manual (TRM) Appendix D, Table 3.6.1.3-1 surveillance requirement SR 3.6.1.3.1 with an "NA" for multiple manually operated primary containment isolation valves, and removes the valves from 0PT-02.2.4A, "Primary Containment Integrity Verification-Containment External." Locks have been placed on each valve and configuration control is being maintained using the associated operating procedure valve lineup checklist.

SUMMARY OF THE EVALUATION:

The position and operation of these manually operated primary containment isolation valves has not been altered. Locks have been placed on affected valves in their currently required position.

PLANT REFERENCES: TRM 2003-04, 0PT-02.2.4A Revision 38, RAINS 03-0600

Changes, Tests, and Experiments

TITLE: Extended Power Uprate High Pressure Turbine Replacement

BRIEF DESCRIPTION:

As part of changes needed to implement Unit 2 EPU, the high pressure turbine rotor has been replaced, the ability of the high pressure and low pressure rotors to operate at 120% original licensed thermal power (OLTP) was evaluated, and the setpoint limits were evaluated and redefined for operation at resultant pressures and flows. The high pressure turbine buckets and diaphragms have been replaced and redesigned with larger flow areas and the turbine first stage pressure switches replaced. In addition, setpoint changes have been implemented to the turbine stop valve closure and turbine control valve fast closure scram bypass trip and low pressure turbine hood spray control system positioner to accommodate the higher backpressure of the high pressure turbine replacement.

SUMMARY OF THE EVALUATION:

Detailed high and low pressure rotor analyses were performed based on the new steam loading and determined that the new rotor is fully suitable for uprated conditions. The new design ensures sufficient pressure control range to control system disturbances at EPU conditions.

PLANT REFERENCES: EC 47903, RAINS 02-1095

Changes, Tests, and Experiments

TITLE: Unit 2 Out of Step Protective Relay

BRIEF DESCRIPTION:

EC 47898 provides the required evaluation and installation and testing instructions for implementation of Out-of-Step Protective Relay on Unit 2. Out-of-Step (OOS) blocking has been installed at both terminals for all of the lines connected at the switchyard for Unit 2. This will prevent the non-faulted lines from opening if an unstable power swing is detected and thus mitigate loss of off-site power issues. An OOS protective scheme can detect an out-of-step condition in the lines or unit step-up transformers and trip the unstable unit(s) as quickly as feasible. Because the OOS tripping protection may be too slow to prevent instability in generators remote from the fault, an additional protective scheme was used to provide faster tripping of the Unit connected to the faulted line. The additional protection will detect the presence of a breaker-failure condition, a double line to ground or three-phase fault, and the distance of the fault from the unit switchyard. It will use these conditions to anticipate that an unstable condition exists for the protected generator. The OOS protective schemes and the additional relay protection will mitigate damage to the unstable generator(s) and disruption to other grid facilities in the service area.

SUMMARY OF THE EVALUATION:

The subject modification will provide additional protection against OOS conditions, thereby improving offsite power reliability and grid stability. Analyses in support of this modification are conservative in nature. Impacts to accidents, anticipated operational occurrences, and safety related equipment described in the UFSAR with regard to frequency, results and consequences are acceptable. The proposed modification will not impact the ability to achieve or maintain safe shutdown.

PLANT REFERENCES: EC 47898, RAINS 02-1146

Changes, Tests, and Experiments

TITLE: Unit 2 Extended Power Uprate

BRIEF DESCRIPTION:

EC 47907 implements the configuration control and design/licensing basis changes needed to implement EPU Unit 2 operation. Setpoints for the main steam high flow trips are changed by this EC.

SUMMARY OF THE EVALUATION:

Detailed evaluations of the reactor, engineered safety features, power conversion, emergency power, support systems, environmental issues, design basis accidents, and previous licensing evaluations were performed. The results of the evaluations determined that EPU implementation does not create a significant challenge to any safety system, nor result in significant radiological environmental effects or non-radiological environmental effects.

PLANT REFERENCES: EC 47907, RAINS 02-1098

Changes, Tests, and Experiments

TITLE: Unit 2 Drywell Equipment Drain Tank Temporary Modification

BRIEF DESCRIPTION:

EC 49294 implements a temporary modification to address a hot fluid leak into the Unit 2 Drywell Equipment Drain (DWED) sump which results in cycling of the DWED system motor operator valves and the sump pump motors. This EC provides temporary alignment/operation of equipment, breakers, annunicators, procedures, timers and relays for the purpose of positioning the DWED system in constant sump recirculation cooling.

SUMMARY OF THE EVALUATION:

The modified configuration enables manual start/stop of the drywell equipment drain sump pumps, and the automatic pump start function on a high level, high-high level, or a high temperature condition. The control room operator will still retain the ability to measure the drywell leakage rate after the implementation of this temporary modification with the Drywell Floor Drain system. The changes made by this temporary modification will not affect any system other than the floor and equipment drain system. The proposed operation of the drywell equipment drain system with the temporary modification affects current settings for the sump fill rate timer and the pump out timer on the floor drain system. However, operating instructions provide reasonable assurance that the intent of the annunciators is still maintained. The containment isolation valve function of this system, which provides the barrier for fission product release, is not affected by this modification. The minimum level in the sump is not changed. The proposed activity does not prevent the ability to measure reactor coolant system operational leakage and calculating total leakage in accordance with Technical Specification (TS) requirements.

PLANT REFERENCES: EC 49294, RAINS 02-0834

Changes, Tests, and Experiments

TITLE: Unit 1 Option III Stability

BRIEF DESCRIPTION:

This EC evaluates the PRNM / Option III Reactor Stability Algorithm Modifications and the changes to the Unit 1 reactor coolant recirculation (RCR) pump speed limiter Nos. 1 and 2. This EC supports installation of the required equipment for the startup of Unit 1.

SUMMARY OF THE EVALUATION:

This EC provides a combination of hardware and firmware to provide improved protection in the event of an instability condition. The potential for a thermal-hydraulic instability event is unchanged by the changes made by this EC to the license bases and defense in depth algorithms. By adjusting the tuning parameters to a more sensitive setting, the probability of detecting and suppressing such an event is enhanced.

PLANT REFERENCES: EC 50098, RAINS 03-1205

Changes, Tests, and Experiments

TITLE:

Reactor Building Ventilation Fan Operation

BRIEF DESCRIPTION:

This change enhances the UFSAR description of the reactor building ventilation system. Specifically, information concerning fan operation was revised to reflect that fan operation variation may be necessary during seasonal/climatic conditions such as colder temperatures or excessive wind conditions.

SUMMARY OF THE EVALUATION:

This change has no effect on the ability of the reactor building ventilation system to perform its intended function, nor will it result in degradation to building negative pressures or internal air flow.

PLANT REFERENCES: 01-FSAR-026, RAINS 01-0812

Changes, Tests, and Experiments

TITLE: Administrative Changes to Nuclear Assessment Section Procedure

BRIEF DESCRIPTION:

Procedure NUA-NGGC-1510, "Nuclear Assessment Process," provides direction for assessment activities conducted by the Nuclear Assessment Section. This change is administrative in nature; however since UA-NGGC-1510 implements controls described in the Quality Assurance Program Description as defined in chapter 17.3 of the UFSAR, the conservative decision was made to complete a 10 CFR 50.59 evaluation for this change.

SUMMARY OF THE EVALUATION:

These changes are purely administrative in nature.

PLANT REFERENCES: NUA-NGGC-1510 Revision 15, RAINS 01-0825

Changes, Tests, and Experiments

TITLE:

Monitoring of 2A Reactor Recirculation System Speed Controller

BRIEF DESCRIPTION:

Multipoint recorders were temporarily installed in parallel with several points associated with the 2A RCR system speed controller to determine the source of sporadic speed steps that have occurred during system demands and the necessary corrective actions to resolve the concern.

SUMMARY OF THE EVALUATION:

The installed recorder is a high impedance voltage sensing device with a low probability of failure due to a short or open circuit. An open circuit will have no affect on speed control function. A short circuit could produce a pump runaway condition in either the upward or downward direction. Positive runaway is limited to 2% since the 2A RCR pump will be verified to be operating at greater than 96% speed. A downward runaway is not expected to enter the exclusion area of thermal hydraulic instability and is fully bounded by analyzed transient scenarios.

PLANT REFERENCES: TCF 01-013, RAINS 01-0585

Changes, Tests, and Experiments

TITLE: Technical Requirements Manual Hoist Interlock Test Frequency

BRIEF DESCRIPTION:

This change clarifies the refueling hoist test requirement frequencies as delineated within TRM 3.25. Specifically, TRM 3.25 was revised to state: "Once within 7 days prior to start of movement of fuel assemblies or control rods...if not performed within the previous 30 days."

SUMMARY OF THE EVALUATION:

This change is consistent with the requirements specified in NUREG-0612, "Control of Heavy Loads at Nuclear Plants," ANSI B30.2-1976, "Overhead and Gantry Cranes," applicable information contained with the UFSAR, and recent industry interpretation.

PLANT REFERENCES: TRM 2002-05, RAINS 02-0993

Changes, Tests, and Experiments

TITLE: Spent Fuel Pool Cleaning Procedure

BRIEF DESCRIPTION:

This change involves development of procedure 0SP-02-002, "Spent Fuel Pool Clean-up Project for Unit 1 and Unit 2." This procedure provides instruction for the handling, processing, packaging, and removal of radioactive waste and irradiated hardware from the fuel pools. In addition, the procedure includes instruction for placing a liner and advanced crusher-shear assembly inside the fuel pool. These assemblies will not be restrained while in the fuel pool.

SUMMARY OF THE EVALUATION:

Analysis demonstrates that a seismic event, a potential exists for the liner to overturn whereas the crusher-shear assembly would slide. The potential for damage to irradiated fuel is minimized by locating the liners away from fuel racks to the maximum extent practical. Spent fuel rack supports will prevent direct impact on irradiated fuel by the sliding crusher-shear assembly. In the event the liner overturns the spent fuel pool liner will prevent the loss of spent fuel pool water despite local damage.

PLANT REFERENCES: 0SP-02-002, RAINS 02-0726

Changes, Tests, and Experiments

TITLE: Unit 1 Reactor Building Temporary Cooling

BRIEF DESCRIPTION:

This change temporarily installs pipe supports and electrical power to support testing the feasibility of using additional cooling to enhance personnel comfort in the Unit 1 reactor building during summer months.

SUMMARY OF THE EVALUATION:

The existing reactor building ventilation system is not adversely impacted by this change. The consequence to safety related systems will not be increased beyond that already analyzed. The new temporary components are not located such that safety related components could be damaged in the event of a failure of the temporary equipment. No flooding potential exists in the event of piping failure.

PLANT REFERENCES: EC 47442, RAINS 02-1107

Changes, Tests, and Experiments

TITLE: Unit 1 Steam Jet Air Ejector Troubleshooting

BRIEF DESCRIPTION:

Troubleshooting of the Unit 1 Steam Jet Air Ejector (SJAE) intercondenser drain system's idle train was performed to establish a small drain path to address system inleakage.

SUMMARY OF THE EVALUATION:

The affected train is completely isolated from the condensate system during the evolution. Draining to the condenser was monitored to ensure prompt implementation of mitigative actions in the event condenser vacuum was affected.

PLANT REFERENCES:

TCF 02-035 and 03-028, Equipment Control 1-EC-02-232, RAINS 02-1823 and 03-1133

Changes, Tests, and Experiments

TITLE: Unit 2 Steam Jet Air Ejector After Condenser Tube Leak Detection

BRIEF DESCRIPTION:

This troubleshooting activity involved the manipulation of a single condensate system valve while monitoring after condenser shell level to assist in detection of a possible condenser tube leak.

SUMMARY OF THE EVALUATION:

The leak detection methodology is non-intrusive. The affected SJAE train was in standby during the troubleshooting evolution. The 2A after condenser level was monitored during the activity to ensure level remained within acceptable limits to protect the system's hydrogen oxygen analyzers and contingency steps established for manual control of after condenser level if needed.

PLANT REFERENCES: TCF 02-030, RAINS 03-1318

Changes, Tests, and Experiments

TITLE: Unit 1 Reactor Building Ventilation Troubleshooting

BRIEF DESCRIPTION:

This troubleshooting activity provided a means of determining the most acceptable method to dampen the reactor building pressure control response to building pressure changes. This testing will support decreasing fan oscillations, which has resulted in main steam line isolation valve (MSIV) pit outlet damper closures and subsequent MSIV pit temperature increases.

SUMMARY OF THE EVALUATION:

To support the troubleshooting effort, temporary controllers were installed in the system to control the vortex dampers. The temporary controllers provided equivalent or better control of building pressure. Control of the vortex dampers in this manner has no affect on the reactor building ventilation isolation valves or the safety-related room coolers, since their functions operate independently of the normal ventilation system controls.

PLANT REFERENCES: TCF 02-013, RAINS 02-0693

Changes, Tests, and Experiments

TITLE: Main Steam Isolation Valve Pit Plug Removal

BRIEF DESCRIPTION:

Attachment 12 to procedure 0OI-01.03, "Non-Routine Activities," was established to administratively control access to the MSIV pit during power operation.

SUMMARY OF THE EVALUATION:

The entry into the MSIV pit for the purpose of performing leak inspection or minor maintenance not affecting the high energy pressure retaining boundaries or associated isolation components and functions is considered acceptable. Any additional risk incurred is minimal. Preemptive identification of potential leaks avoids a potential challenge to safety systems and fission product barriers by reducing the possibility of a full MSIV closure transient.

PLANT REFERENCES: 0OI-01.03 Revision 16, RAINS 03-1206

Changes, Tests, and Experiments

TITLE: Reactor Protection System Electrical Protection Assembly Testing

BRIEF DESCRIPTION:

TS Bases Section 3.3.8.2 was clarified with respect to Reactor Protection System (RPS) Electrical Protective Assemble (EPA) breaker testing during power operations; aligning the TS Bases with the TS.

SUMMARY OF THE EVALUATION:

Existing administrative processes for on-line scheduling of activities, along with the outstanding EPA breaker setpoint performance and minimal spurious signals, result in adequate basis for performing on-line testing of the EPA breakers.

PLANT REFERENCES: TSB 2002-01, RAINS 02-0921

Changes, Tests, and Experiments

TITLE: Steam Jet Air Ejector Train Annunciators Disabled

BRIEF DESCRIPTION:

This activity involves the disabling of a series of annunciators that provide indication for the idle SJAE trains in Units 1 and 2. The annunciator logic configuration for the trains was originally designed to operate with two SJAE trains in half-load operation. The normal operating configuration for SJAEs has now been changed to single train in full-load operation. As such, a series of the idle train's annunciators provide erroneous readings due to there being no process flow through that train.

SUMMARY OF THE EVALUATION:

The affected annunciators do not directly affect system operation in any way and only provide information to the operators. Moreover, the train with the disabled annunciators will be idle at the time of their disabling.

PLANT REFERENCES: Action Request 61303, RAINS 02-1072

Changes, Tests, and Experiments

TITLE:

Operation without Main Turbine Backup Pressure Regulators

BRIEF DESCRIPTION:

This activity involves incorporation of additional guidance within operating procedures regarding the impacts of operation with the main turbine backup pressure regulator inoperable.

SUMMARY OF THE EVALUATION:

This change clearly denotes that operation at reactor thermal power (RTP) levels between 23% for Unit 1 and 25% for Unit 2 and 90% without a backup pressure regulator may be an unanalyzed condition. Operation at greater than 90% RTP without a backup regulator is bounded by existing analyses. Operation at low power (i.e., less than 23% RTP for Unit 1 and 25% RTP for Unit 2), involves a large inherent margin that ensures the Minimal Critical Power Ratio is not exceeded.

PLANT REFERENCES: 0GP-03, 0GP-04, 0GP-05, 0GP-12, RAINS 02-1791

Changes, Tests, and Experiments

TITLE: Steam Dryer Load Analysis Data Collection

BRIEF DESCRIPTION:

This troubleshooting activity involves the monitoring of reactor vessel and main steam line pressures by connecting temporary pressure transducers to calculate the dynamic loading on the reactor vessel dryer.

SUMMARY OF THE EVALUATION:

This activity incorporates appropriate equipment and administrative controls to ensure minimal risk of system perturbations. Installed components are not impacted by the installation of the temporary equipment. The robust design of the temporary equipment resolves any pressure boundary concerns. Component manipulations were performed by qualified personnel. Any potential leakage or spurious actuation event is bounded by analyzed events.

PLANT REFERENCES: TCF 04-001, RAINS 04-0209

Changes, Tests, and Experiments

TITLE: Technical Requirements Manual Administrative Changes

BRIEF DESCRIPTION:

Various administrative changes to the TRM were incorporated in Revisions 25 for Unit 1 and Revision 21 for Unit 2. At the time of this change, the process for changes to the TRM required a complete evaluation in accordance with 10 CFR 50.59.

SUMMARY OF THE EVALUATION:

These changes are purely administrative in nature, do not reduce the level of control applied to the TRM, and represent no physical change to the facility or operation of facility equipment.

PLANT REFERENCES: TRM 2003-05, RAINS 03-0658

Commitment Changes

ORIGINATING DOCUMENT:

The NRC issued Bulletin 80-13, "Cracking in Core Spray Spargers," by letter dated May 12, 1980. A report of the examination results was required to be submitted within 30 days of completion of the examinations. Carolina Power & Light Company, now doing business as Progress Energy Carolinas, Inc. (PEC), began reporting examination results, beginning with Brunswick Steam Electric Plant (BSEP), Unit 2, by letter dated July 7, 1982.

SUBJECT OF CHANGE:

Revise the commitment for inspection of reactor pressure vessel internal core spray piping and spargers.

ORIGINAL COMMITMENT:

In response to Bulletin 80-13, PEC committed to inspect reactor pressure vessel internal core spray piping and spargers in accordance with Bulletin 80-13.

REVISED COMMITMENT:

Future inspections of the in-vessel core spray piping and spargers will be performed in accordance with BWRVIP-18, "BWR Core Spray Internals Inspection and Flaw Evaluation Guidelines," as approved by the NRC.

BASIS:

Prior to issuance of BWRVIP-18, inspections of the BSEP in-vessel core spray piping and spargers were conducted in accordance with the requirements of Bulletin 80-13. The NRC closed Bulletin 80-13 on the basis that inspections be continued in accordance with the requirements of Bulletin 80-13. The Executive Summary of BWRVIP-18 states that "...for BWRVIP members, these guidelines can be followed in the place of prior GE SILs (Service Information Letters) and, when approved by the regulator, in place of the requirements of IE Bulletin 80-13." The NRC approved BWRVIP-18 in its Final Safety Evaluation Report dated December 2, 1999.

Commitment Changes

ORIGINATING DOCUMENT:

NRC Inspection Report Nos. 50-325/95-13 and 50-325/95-14

SUBJECT OF CHANGE:

Delete the project management procedure, ADM-NGGC-0103.

ORIGINAL COMMITMENT:

PEC committed to proceduralizing the Project Management Manual in a Reply to a Notice of Violation NRC Inspection Report Nos. 50-325/95-13 and 50-325/95-14 (i.e., Serial BSEP 95-0519).

REVISED COMMITMENT:

Delete the commitment requiring a proceduralization of a Project Management Manual.

BASIS:

The project management manual was considered an enhancement to the engineering process and is not specifically required by regulation. 10 CFR 50 Appendix B, Criterion III, Design Control, requires, in part, that measures be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications and instructions. Criterion III also requires, in part, that design control measures shall provide for verifying or checking the adequacy of design such as by design reviews or by the performance of a suitable testing program. The requirements of 10 CFR 50 Appendix B, Criterion III, are controlled via existing engineering processes. These processes are controlled by procedures such as EGR-NGGC-0005, "Engineering Change," EGR-NGGC-0003, "Design Review Requirements," and EGR-NGGC-0011, "Engineering Products Review."

Commitment Changes

ORIGINATING DOCUMENT:

NRC Bulletin 79-18, "Audibility Problems Encountered on Evacuation of Personnel from High-Noise Areas," described emergency evacuation problems encountered at a nuclear facility. The bulletin required investigation and corrective action to assure that areas identified as inaudible will receive adequate audible/visual evacuation signals. In addition, in areas where adequate audible/visual evacuation signals cannot be assured by hardware changes, the bulletin required a determination of what additional administrative measures were necessary to assure personnel evacuation.

SUBJECT OF CHANGE:

Use adequate administrative means in lieu of a public address (PA) system for communication of evacuation within the Units primary containments.

ORIGINAL COMMITMENT:

In response to Bulletin 79-18 PEC committed to the following:

> An investigation was conducted to determine all areas at the Brunswick Steam Electric Plant where audibility of the various alarms (fire, evacuation, etc.,) was degraded... The inoperable speakers were repaired, and locations were identified in an Engineering Work Request, where additional speakers are to be installed. Present plans are to install these additional speakers at designated areas such as indicated on Enclosure 1...(Enclosure 1 listed the drywell, reactor building-Unit1).

REVISED COMMITMENT:

PEC will use adequate administrative means, (i.e., air horn or suitable substitute and the current practice of personnel accountability) to assure adequate communication of evacuation where needed.

BASIS:

PEC initially met the requirements of Bulletin 79-18 by installing drywell PAs. Subsequently, it has been determined that drywell conditions make PA maintenance impractical. Accordingly, additional administrative measures, such as air horns and personnel accountability assure personnel evacuation as allowed by the Bulletin 79-18.