Progress Energy

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10 CFR 50.59(d)(2)

United States Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, DC 20555

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
REPORT OF CHANGES PURSUANT TO 10 CFR 50.59

Ladies and Gentlemen:

In accordance with 10 CFR 50.59(d)(2), Progress Energy Carolinas, Inc. hereby submits the report of "Changes, Tests and Experiments," for the Harris Nuclear Plant. The report provides a brief description of changes to the facility and a summary of the safety evaluation for those items implemented under 10 CFR 50.59 between January 4, 2002 and May 18, 2003 (end of refueling outage 11).

If you have any questions regarding this submittal, please contact me at (919) 362-3137.

Sincerely,

John R. Caves

J. R. Caves
Supervisor, Licensing/Regulatory Programs
Harris Nuclear Plant

JRC/mgw

Enclosure

c: Mr. R. A. Musser (NRC Senior Resident Inspector, HNP)
Mr. C. P. Patel (NRR Project Manager, HNP)
Mr. L. A. Reyes (NRC Regional Administrator, Region II)

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IE47

SE # 01-0013

Title: OST-9029T, Temporary procedure for testing PIC card 0526 in PIC Cabinet 6

Description:
PIC card 0526 was replaced for corrective maintenance. Procedure OST-9029T performs testing to ensure that the associated controller manual auto stations properly transfer to and from the Auxiliary Control Panel (ACP). The transfer is accomplished using a switched jumper, versus the transfer switches to limit the amount of components that are transferred.

Safety Summary:
This temporary procedure was developed to perform post-maintenance testing of PIC card 0526. It accomplishes this by installing a switched jumper in PIC-06 and transferring only the manual auto stations affected by the card. While use of the jumper is different than the assumed method of transfer using the transfer relays, this method does not introduce any mechanism which would cause an accident or malfunction of equipment other than the failures already assumed. None of the components affected by this test are credited for safe shutdown, so the ability to shutdown and maintain the plant is unchanged. This test will transfer the following Manual/Auto stations: FK-479.1 Feed Reg Bypass flow control A, TK-381A1 Boron Thermal Recovery System demin inlet temperature control, and PK-145.1 Letdown Pressure controller. During test performance, a licensed Senior Reactor Operator is stationed at the ACP with continuous communications with the Main Control Room during the time of the transfer. The level of monitoring for the affected components is unchanged or increased. If the switch were to fail open, the transfer would be unsuccessful, and control would remain at the Main Control Board. This activity does not increase the probability or consequences of accidents or malfunctions of equipment previously evaluated, does not introduce a different type of accident or malfunction of equipment, and does not reduce any margin of safety. Therefore, the change does not involve an unreviewed safety question.

SE # 01-2238

Title: Engineering Service Request (ESR) 99-00226 Rev. 0, Evaluation of fire doors for inconsistent fire door designations on plant documents.

Description:

The activity is an evaluation to identify specifically which plant doors are located in rated fire barriers credited for NUREG 0800 fire safe shut down. As a result, plant surveillance procedures, Equipment Data Base, and the Final Safety Analysis Report (FSAR) are revised to identify the credited fire doors.

Safety Summary:

The FSAR describes and shows on figures, the rated fire barriers in the plant. Both the FSAR descriptions and figures are revised to show the changes to fire barriers in the Reactor Auxiliary Building, Turbine Building, Fuel Handling Building, Waste Processing Building, and Diesel Fuel Oil Storage Building. The Safe Shutdown Analysis in Case of Fire (calculation E-5525) which is incorporated into the FSAR by reference is also revised to reflect the changes in the fire barriers. The procedures affected by this evaluation are not discussed in the FSAR or used as input for any analysis in the FSAR and no special tests or experiments are conducted as part of this ESR. The accident under consideration is a design basis fire. No new ignition sources are introduced, no new types of fire hazards are introduced, and the fire protection program administrative controls for ignition sources and hot work are not changed. Therefore the ignition frequency for fires in the affected areas is not affected. This change does not increase the consequences of an accident previously evaluated in the FSAR or increase the consequences of a malfunction of an Structure, System or Component (SSC) important to safety previously evaluated, nor introduce a different type of accident or malfunction of equipment, and does not reduce the margin of safety as defined in the basis for any Technical Specifications. This change does not result in an unreviewed safety question. This change does not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire. Therefore, a license amendment is not involved.

SE # 02-0152

Title: Engineering Service Request (ESR) 02-00005 Rev. 2, Reactor Coolant Pump (RCP) "A" Standpipe Low and High Level Alarms to be disabled under a temporary modification

Description:
The activity is an evaluation to temporarily remove the high and low level alarms on the Main Control Board for the A, B, and C reactor coolant pumps (RCP) standpipes. The low and high level RCP standpipe switches have proved to be unreliable and the faulty alarms are a distraction to the control room operators.

Safety Summary:
The Main Control Board annunciators are used to provide alarm or indication of abnormal plant conditions to the control room operators. The annunciator alarms are locked-in for the A, B, and C RCP standpipe (i.e., low level condition), and the associated high level switches are not functioning properly, indicating faulty switches. Disabling the input to the standpipe level alarms until they can be repaired during the next refueling outage will not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the Final Safety Analysis Report (FSAR). Operations personnel have other existing procedural guidance, which is used when the alarms are defeated to maintain proper level control of the RCP A, B and C standpipes between the high and low alarm level alarm set points. The A RCP standpipe will be filled by manual operation. Disabling the faulty alarm inputs will not cause an increase in the likelihood of occurrence of a malfunction of a Structure, System or Component (SSC) important to safety previously evaluated in the FSAR. The annunciator system is not an accident mitigating system, and is not required for safe shutdown. Disabling the associated annunciator circuits will not affect any fission product barriers, and will not increase fission product release during an accident. Therefore, the proposed activity does not result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR or in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated, nor introduce a different type of accident or malfunction of equipment, does not result in exceeding a design basis limit for a fission product barrier, or a departure from a method of evaluation described in the FSAR. Therefore, a license amendment is not involved.

Title: Engineering Change (EC) 51836, Revise Final Safety Analysis Report (FSAR) and Design Basis Document (DBD) to allow the spent fuel pool heat loads to be calculated with the ORIGEN2 computer code

Description:
EC 51836 changes the FSAR description of the methodology used in calculating the heat load on the spent fuel pool cooling system. The FSAR currently describes use of NRC Branch Technical Position (BTP) 9-2 as modified by FSAR Section 9.1.3.1.g. EC 51836 changes the FSAR described methodology to the ORIGEN2 computer code.

Safety Summary:

The proposed activity involves a change in the methodology as reflected in the FSAR to ORIGEN2 for the calculation of the thermal decay heat load from spent fuel on the Spent Fuel Pool Cooling and Cleanup System (FPCCS). The method for analyzing the thermal energy to be removed by the spent fuel cooling system is specifically listed in FSAR Section 9.1.3. The method currently listed is NRC BTP ASB 9-2. The proposed activity does not result in a departure from a method of evaluation. The proposed activity deals specifically with a change to the ORIGEN2 method for decay heat load calculations in spent fuel pools that has been previously reviewed and approved by the NRC for other licensees. The terms, conditions and limitations of the use of ORIGEN2 from the previous approved NRC Safety Evaluation Reports have been reviewed. It has been determined that the application at Harris Plant, as described in EC 51836, is within the terms and conditions of the referenced approved applications. The qualifications of Progress Energy to use the method have been reviewed and found satisfactory. Therefore this activity does not require NRC review and approval. The proposed activity does not have any impact on the frequency of occurrences of an accident previously analyzed in the FSAR. The FPCCS is designed to maintain the spent fuel pool below a specified temperature given the design heat loads. Since neither the maximum allowed temperature nor any other operating parameter of a Structure, System or Component (SSC) is being changed by the proposed activity, the occurrence of a malfunction of an SSC important to safety is not affected. The change in the methodology for calculating decay heat does not have an impact on the consequences of a fuel handling accident or any other accident evaluated in the FSAR. The design heat loads are not being reduced by the proposed activity and therefore the existing FSAR design analysis of the FPCCS remains unchanged. The proposed activity does not change the configuration of any SSC important to safety. Therefore, the proposed activity does not create the possibility of an accident of a different type than any previously evaluated in the FSAR. The proposed activity does not create the possibility of a malfunction of a SSC important to safety with a different result than any previously evaluated in the FSAR. The proposed activity does not affect the three fission product barriers (fuel, reactor coolant system pressure boundary or containment boundary). It has been determined that a License Amendment is not required.

Title: Justification for use of the new fuel racks A1, A2, A3 and A4 in accordance with FMP-106 Revision 14 and EMF-93-139, Revision 1.

Description:
The activity is being performed as part of an extent of condition for Nonconformance Report AR86705. A deficiency was noted in the 50.59 evaluation performed for the Cycle 6 reload(the transition from Westinghouse to Siemens fuel) with respect to the storage of Siemens 17x17 fuel in the dry storage racks in the new fuel inspection pit. This 50.59 evaluation is being performed to address that deficiency and document the acceptability of the new fuel storage racks containing Siemens (Framatome) fuel.

Safety Summary:
The implementing activity is the formal justification for use of the new fuel racks A1, A2, A3 and A4 in accordance with FMP-106 revision 14 and EMF-93-139, revision 1, up to a maximum enrichment of 5.0 weight percent U-235. The criticality analysis for the new fuel pit has been documented in EMF-93-139 and found to meet the regulatory requirements provided in SRP 9.1.1, 10 CFR 50 Appendix A, GDC 62, and Regulatory Guide 1.13. The implementing procedure FMP-106 has the available new fuel locations consistent with the analysis in racks A1, A2, A3, and A4. The proposed activity will permit storage of new fuel in the new fuel storage racks A1, A2, A3, and A4. A criticality analysis has been performed and confirms that the FSAR described design function of the racks to maintain the k-eff of the fuel assembly array to less than the regulatory requirement of 0.95 has been satisfied. Therefore, the proposed activity does not does not result in an increase in the frequency of occurrence of an accident previously evaluated in the FSAR. The criticality analysis is not an accident initiator and does not affect the fuel handling procedures governing fuel movement with the exception of the determination of acceptable locations to place the fresh fuel. The proposed activity does not result in an increase in the likelihood of occurrence of a malfunction of a Structure, System or Component (SSC) important to safety previously evaluated in the FSAR. The criticality analysis confirms that the SSC performs as intended when the new fuel is loaded in accordance with the analysis, as controlled by FMP-106. The criticality analysis does not affect the fuel handling procedures governing fuel movement with the exception of the determination of acceptable locations to place the fresh fuel. Since the criticality of fuel assemblies outside the reactor (in new fuel Racks A1, A2, A3 and A4) has been demonstrated in the criticality analysis to be precluded, there is no increase in the consequences of an accident previously evaluated in the FSAR. There is no increase in the consequence of a malfunction of an SSC important to safety previously evaluated in the FSAR. Since adminstrative controls have been in place for spent fuel racks, similar adminstrative controls implemented for fresh fuel racks do not create the possibility for an accident of a different type than previously evaluated in the FSAR. The proposed activity does not result in exceeding a design basis limit for a fission product barrier or a departure from a method of evaluation described in the FSAR. Therefore, a license amendment is not involved.