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RECIP.NAME RECIPIENT AFFILIATION
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SUBJECT: Application for amend to license NPF-63,requesting rev to chemist y data (nickel content) in TS 3/4.4.9 re pressure/ temp limits.

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CP&L

Carolina Power & Light Company
PO Box 165
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William R. Robinson
Vice President
Harris Nuclear Plant

SERIAL: HNP-96-206
10 CFR 50.90
10 CFR 50 App. G
10 CFR 50 App. H

DEC 30 1996

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
REQUEST FOR LICENSE AMENDMENT
RCS PRESSURE/TEMPERATURE LIMITS

Dear Sir or Madam:

In accordance with the Code of Federal Regulations, Title 10, Part 50.90, Carolina Power & Light Company (CP&L) requests a revision to the Technical Specifications (TS) for the Harris Nuclear Plant (HNP). The proposed amendment to Technical Specification (TS) 3/4.4.9, "Pressure/Temperature Limits," revises chemistry data (nickel content) shown on TS Figures 3.4-2 and 3.4-3. In addition, the associated Bases 3/4.4.9, is revised to reflect changes to chemistry and material properties and changes to comply with recent NRC rule changes to 10 CFR 50, Appendix G.

Also attached to this letter is a corresponding revision to the Reactor Vessel Surveillance Capsule Report, originally submitted on April 2, 1992, in compliance with 10 CFR 50, Appendix H. This license amendment request and revision to the Reactor Vessel Surveillance Capsule Report are being submitted in accordance with commitments contained in the HNP response to NRC Generic Letter 92-01, Revision 1, Supplement 1, dated November 16, 1995.

Enclosure 1 provides a description of the proposed changes and the basis for the changes.

Enclosure 2 details, in accordance with 10 CFR 50.91(a), the basis for the Company's determination that the proposed changes do not involve a significant hazards consideration.

Enclosure 3 provides an environmental evaluation which demonstrates that the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental assessment needs to be prepared in connection with the issuance of the amendment.

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Enclosure 4 provides page change instructions for incorporating the proposed revisions.

Enclosure 5 provides the proposed Technical Specification pages.

Enclosure 6 contains the Reactor Vessel Surveillance Capsule Report Revision.

CP&L requests that the proposed amendment be issued such that implementation will occur within 60 days of issuance to allow time for procedure revision and orderly incorporation into copies of the Technical Specifications.

Please refer any questions regarding this submittal to Ms. D. B. Alexander at (919) 362-3190.

Sincerely,

W R Robinson

W. R. Robinson

JHE/jhe
Enclosures:
1. Basis for Change Request
2. 10 CFR 50.92 Evaluation
3. Environmental Considerations
4. Page Change Instructions
5. Technical Specification Pages
6. Reactor Vessel Surveillance Capsule Report Revisions

W. R. Robinson, having been first duly sworn, did depose and say that the information contained herein is true and correct to the best of his information, knowledge and belief; and the sources of his information are employees, contractors, and agents of Carolina Power & Light Company.

Darlene S. Yarborough
Notary (Seal)

My commission expires: 2-6-2000

DARLENE S. YARBOROUGH NOTARY *** PUBLIC HARNETT COUNTY, NC

c: Mr. J. B. Brady, NRC Sr. Resident Inspector
Mr. Dayne H. Brown, N.C. DEHNR
Mr. S. D. Ebneter, NRC Regional Administrator
Mr. N. B. Le, NRC Project Manager

bc:

Ms. P. B. Brannan
Mr. H. K. Chernoff (RNP)
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Mr. J. W. Donahue
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Mr. G. A. Rolfson
Mr. R. S. Stancil
Mr. M. A. Turkal (BNP)
Mr. T. D. Walt
Nuclear Records
Harris Licensing File
File: H-X-0511

ENCLOSURE 1

SHEARON HARRIS NUCLEAR POWER PLANT
NRC DOCKET NO. 50-400/LICENSE NO. NPF-63
REQUEST FOR LICENSE AMENDMENT
RCS PRESSURE/TEMPERATURE LIMITS

BASIS FOR CHANGE REQUEST

Background

Title 10 of the Code of Federal Regulations, Part 50, Appendix A , General Design Criterion 31, "Fracture Prevention of Reactor Coolant Pressure Boundary", requires that the reactor coolant pressure boundary be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions: (1) the boundary behaves in a nonbrittle manner, and (2) the probability of rapidly propagating fracture is minimized.

Appendix G to 10 CFR 50, "Fracture Toughness Requirements", describes specific requirements for fracture toughness and reactor vessel operation to meet the 10 CFR 50.60 criteria regarding prevention of brittle fracture. In addition, Appendix G requires changes in fracture toughness of reactor vessel materials caused by neutron radiation throughout the service life of nuclear reactors to be considered in the limits on operation. Regulatory Guide 1.99 contains procedures for calculating the effects of neutron radiation embrittlement of low-alloy steels used for light water-cooled reactor vessels.

In accordance with 10 CFR 50.36(c)(2), limiting conditions for operation are to be included in a plant's Technical Specification (TS). Technical Specifications 3.4.9.1 and 3.4.9.2, "REACTOR COOLANT SYSTEM PRESSURE/TEMPERATURE LIMITS", provide Reactor Coolant System (RCS) Pressure-Temperature Limits to protect the reactor pressure vessel from brittle fracture by clearly separating the region of normal operations from the region where the reactor vessel may be subject to brittle fracture. The current RCS Pressure-Temperature Limitations for the Harris Nuclear Plant (HNP) were developed in accordance with 10 CFR 50 Appendix G criteria and the calculative procedure for determining the adjusted reference temperature in Regulatory Guide 1.99, Revision 2, for a predicted reactor vessel neutron irradiation equivalent to eleven Effective Full Power Years (EFPY). These Pressure-Temperature Limits were approved by the NRC on August 20, 1993, as Amendment 38 to the Operating License.

NRC Generic Letter 92-01, Revision 1, "Reactor Vessel Structural Integrity, 10 CFR 50.54 (f)," requested licensees to provide the NRC with specific information relative to reactor vessel integrity. Carolina Power & Light Company (CP&L) provided a response for HNP in a letter dated July 6, 1992. By letter dated May 13, 1994, the NRC requested CP&L to verify certain information contained in the NRC's Reactor Vessel Integrity Database (RVID) relative to HNP Pressurized Thermal Shock (PTS) and Upper Shelf energy (USE) parameters. HNP provided a

response in a letter dated June 10, 1994. The NRC issued Supplement 1 to Generic Letter 92-01, Revision 1, dated May 19, 1995. This Supplement requested licensees to identify, collect and report any new data pertinent to the analysis of structural integrity and to assess the impact of that data on their Reactor Pressure Vessel's (RPV's) integrity analyses relative to the requirements of 10 CFR 50.60, 10 CFR 50.61 and 10 CFR Part 50 Appendices G and H. These requirements relate to PTS, USE, Pressure-Temperature (P-T) Limits and Low Temperature Overpressure Protection System (LTOPS) setpoints. HNP provided a 90-day response to Part 1 of GL 92-01, Revision 1, Supplement 1, in a letter dated August 17, 1995 and a 6-month response to Parts 2, 3 & 4 of GL 92-01, Revision 1, Supplement 1, in a letter dated November 16, 1995. The NRC issued a closeout letter for GL 92-01, Revision 1, Supplement 1 in regard to HNP, dated August 7, 1996.

In response to GL 92-01, Revision 1, Supplement 1, HNP reviewed available reactor pressure vessel beltline material data and identified plants possessing the same beltline weld heats as those contained within the HNP vessel. Also, CP&L participated in a cooperative data sharing activity with these plants and the Westinghouse Owners Group in an effort to establish "best estimate chemistry" for the beltline materials in the HNP vessel. As a result of the assessment, the best estimate chemistry has been revised for some of the beltline materials; and there were some minor material property changes (T_{NDT} and USE) regarding weld heat 5P6771. The response also indicated that these changes did not adversely impact reactor vessel integrity, however, some documentation changes were necessary, specifically to the chemistry data contained in the inset to Technical Specification Figures 3.4-2 and 3.4-3, Reactor Coolant System Cooldown and Heatup Limitations. Also, the response indicated that these changes would be submitted to the NRC during 1996.

A Rule change to 10 CFR 50, Appendix G was published in the Federal Register on December 19, 1995 and was made effective January 18, 1996. In part, this amended rule described the conditions pertaining to In-Service Leak & Hydrotests (ISLH) and changed the American Society of Mechanical Engineers (ASME) Code Section to be used for the methodology of developing the Pressure-Temperature Limits referred to above.

The purpose of this Technical Specification Change Request is to revise Figures 3.4-2 and 3.4-3 as stated above and to implement the chemistry and material property changes with respect to the Technical Specification Bases. Further, as a result of Rule changes related to 10 CFR 50, Appendix G, effective January 18, 1996, a revision to the Bases of the HNP Technical Specifications is being implemented to comply with the new Rule changes.

Proposed Change

This change revises the chemistry data in the inset to TS Figures 3.4-2 and 3.4-3 (TS 3/4.4.9). Specifically, the nickel content for the controlling material plate of the reactor vessel beltline region, A9153-1, is revised from 0.45% to 0.46%. In addition, the proposed amendment revises the associated Bases, to reflect: (i) chemistry data changes (nickel and copper) for reactor vessel

beltline materials; (ii) dropweight temperature (T_{NDT}) and Upper Shelf Energy (USE) changes for the reactor vessel circumferential weld 5P6771; (iii) the provision to perform In-Service Leak & Hydrotests (ISLH) using the ISLH Pressure-Temperature Limits whenever fuel is in the reactor vessel; (iv) to require that ISLH tests required by the ASME Code to be completed before the core is made critical and (v) to revise the reference for the Pressure-Temperature Limits requirements from ASME Section III, Appendix G to ASME Section XI, Appendix G.

Basis

As indicated in the November 16, 1995, letter to the NRC, the changes in the copper and nickel values for the reactor vessel beltline materials were due to an assessment of the best estimate chemistry made in response to NRC Generic Letter 92-01, Revision 1, Supplement 1. For the affected materials, the changes reflect a reduction in copper and nickel content, except for base metal plates A9153-1, C9924-1 and C9924-2 which experienced a slight increase in nickel content.

The best estimate chemistry for controlling plate A9153-1 has been determined to be 0.09% copper and 0.46% nickel resulting in a chemistry factor of 58 (Table 2 - Regulatory Guide 1.99, Revision 2). Although this is a very slight increase in the current nickel content value, (0.45%), the chemistry factor is not affected. Therefore the Pressure-Temperature Limits in Figures 3.4-2 and 3.4-3, which are based upon the controlling plate material, A9153-1, remain unaffected by the slight increase in nickel content. Since the Pressure-Temperature Limits remain unaffected, there is no adverse impact on the integrity of the Reactor Coolant System by the proposed changes. However, the chemistry data in the inset to TS Figures 3.4-2 and 3.4-3 and in the BASES Table B 3/4.4-1 must be revised.

None of the remaining base metal plate nickel increases (for plates C9924-1 and C9924-2) affected the chemistry factors used in the pressurized thermal shock reference temperature (RT_{PTS}) or in the nil-ductility transition reference temperature (RT_{NDT}) determination. For reactor vessel plate C9924-1, the best estimate chemistry was determined to be 0.08% copper and 0.47% nickel representing a very slight increase in nickel content over the value, 0.45%, currently stated in the BASES Table B 3/4.4-1. However, the chemistry factor remains the same at 51. For reactor vessel plate C9924-2, the best estimate chemistry was determined to be 0.08% copper and 0.47% nickel representing a very slight increase in nickel content over the value, 0.44%, currently stated in the BASES Table B 3/4.4-1. However, the chemistry factor remains the same at 51. Those beltline materials which experienced a reduction in copper or nickel content resulted in a chemistry factor that remained the same or which could have been reduced. For example, for reactor vessel plate B4197-2, the best estimate chemistry has been determined to be 0.09% copper and 0.50% nickel representing a slight reduction in copper content over the value, 0.10%, currently stated in the BASES Table B 3/4.4-1. For the reactor vessel intermediate and lower axial (longitudinal) welds, 4P4784, the best estimate chemistry has been determined to be 0.05% copper and 0.91% nickel representing a slight reduction in copper content over the value, 0.06%, currently stated in the BASES Table B 3/4.4-1. For the reactor vessel circumferential (girth) weld, 5P6771, the best estimate chemistry has been determined to be 0.03% copper and 0.94% nickel representing a slight reduction in the copper and nickel contents over the values, 0.04% & 0.95% respectively, currently stated in the BASES Table B 3/4.4-1. However, for the reasons stated in the November 16, 1995, letter, CP&L has elected at this time to retain the existing docketed chemistry factors as previously reported to you. As stated above, the best estimate copper content for reactor vessel beltline materials, plate B4197-2, weld 4P4784 and weld 5P6771 were reduced slightly. A lower copper content results in a smaller percentage reduction in Upper Shelf Energy (USE) when using the Regulatory Guide 1.99, Revision 2, Position 1.2 method, which is the method applied to the HNP predicted End-Of-Life (EOL) USE. However, the percentage reduction in USE due to the reduction in copper content for these beltline materials was conservatively maintained the same, (i.e. unchanged). Since the chemistry data for the reactor vessel beltline materials is contained in the BASES Table B 3/4.4-1, it must therefore be revised.

CP&L's November 16, 1995 letter stated that a review of newly acquired and available data also indicated a minor change to material properties with respect to weld material heat number 5P6771. This weld material was used in the reactor vessel circumferential weld joining the intermediate and lower shell base metal plates in the beltline region. The dropweight temperature, T_{NDT}, has been determined to be -80°F based on unirradiated surveillance weldment test data, versus -20°F currently stated in the BASES Table B 3/4.4-1. However, this does not impact the initial or unirradiated nil-ductility transition reference temperature, RT_{NDT}, as stated in BASES Table B 3/4.4-1 at -20°F, since it is now based on the surveillance weldment temperature for the Charpy 50 ft-lb value, less 60°F, which results in the same RT_{NDT} value, -20°F. Further, based on the additional test data, the initial or unirradiated USE for the reactor vessel circumferential weld has been determined to be 80 ft-lb, versus 88 ft-lb currently stated in the BASES Table B 3/4.4-1. The initial USE remains above the 75 ft-lbs value prescribed in 10 CFR 50, Appendix G, paragraph IV.A.1. Although this adversely affects the irradiated EOL USE, the value will remain greater than the 50 ft-lb limit prescribed by 10 CFR 50, Appendix G, paragraph IV.A.1. Specifically, the EOL USE is predicted to be 60 ft-lbs at the inside surface and 62 ft-lbs at the quarter thickness (T/4) location. This prediction is based on the Regulatory Guide 1.99, Revision 2, Position 1.2 method, using a conservative percentage reduction in USE, as described above. This weld material is included in the surveillance capsule program, therefore it is expected that the percent reduction in irradiated USE would not be as great if the benefit of the surveillance capsule results were applied as allowed by RG 1.99, Revision 2, Position 2.2. Since the T_{NDT} and USE for the circumferential weld material heat number 5P6771 are contained in the BASES Table B 3/4.4-1, they must therefore be revised.

The existing BASES 3/4.4.9 only allows ISLH testing using the ISLH Pressure-Temperature Limits, provided fuel is removed from the reactor vessel. ISLH testing with fuel in the reactor vessel is allowed using the normal Pressure-Temperature Limits. This was implemented as part of a change to the BASES 3/4.4.9 in Amendment 38 to the Operating Licence proposed in CP&L letter dated February 26, 1993. The change was made to provide consistency with the recommendations of Welding Research Council Bulletin 175, the source input document for the

ASME Appendix G Pressure-Temperature Limits criteria/methodology, in the absence of clear guidance or criteria in 10 CFR 50, Appendix G. As a result of the amended Rule for 10 CFR 50, Appendix G, effective January 18, 1996, the rule clarified, as described in paragraph IV.A.2 and Table 1, the acceptability of allowing ISLH tests to be performed with fuel in the reactor vessel using the ISLH Pressure-Temperature Limits. Therefore the proposed change to BASES 3/4.4.9 in this submittal does not conflict with any regulation. Also, the amended Rule for 10 CFR 50, Appendix G, clarified, as described in paragraph IV.A.2.d, that any ISLH tests required by ASME Section XI must be completed prior to allowing the core to go critical. This is not explicitly stated in the existing BASES 3/4.4.9, therefore the proposed change to BASES 3/4.4.9 makes this clear.

In addition, the amended Rule for 10 CFR 50, Appendix G, revised the particular Section of the ASME Code to be used for the development of the Pressure-Temperature Limits. Specifically, the change referenced ASME Section XI, Appendix G versus ASME Section III, Appendix G currently referenced in BASES 3/4.4.9. The NRC has stated in the Federal Register Notice for the amended Rule that changing of the reference from ASME Section III, Appendix G to ASME Section XI, Appendix G has no impact because the requirements in the Appendices are identical. Therefore, BASES Table B 3/4.4-1, is being revised to be consistent with the new Rule.

Conclusions

The previously reported chemistry factors for the reactor vessel beltline materials are not adversely affected by the minor changes in chemistry composition described above. With this consideration and the fact that the initial RT_{NDT} for weld heat 5P6771 has not changed due to the reduction in T_{NDT} value, there is no impact to previously reported values for the adjusted nil-ductility transition reference temperature (ART_{NDT}) or pressurized thermal shock reference temperature (RT_{PTS}). Consequently, there is no adverse impact on the operating Pressure-Temperature (P-T) Limits described in the HNP Technical Specifications, Low Temperature Overpressure Protection System (LTOPS) setpoints, or Emergency Operating procedures (EOP's). The reduction in USE at EOL for weld 5P6771 remains above the criterion specified in 10 CFR 50, Appendix G and therefore does not adversely affect reactor vessel integrity. Other beltline material USE values at EOL remain unaffected. The proposed ISLH test condition is in compliance with the amended Rule, 10 CFR 50, Appendix G, and is therefore acceptable.

ENCLOSURE 2

SHEARON HARRIS NUCLEAR POWER PLANT
NRC DOCKET NO. 50-400/LICENSE NO. NPF-63
REQUEST FOR LICENSE AMENDMENT
RCS PRESSURE/TEMPERATURE LIMITS

10 CFR 50.92 EVALUATION

The Commission has provided standards in 10 CFR 50.92(c) for determining whether a significant hazards consideration exists. A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety. Carolina Power & Light Company has reviewed this proposed license amendment request and determined that its adoption would not involve a significant hazards determination. The bases for this determination are as follows:

Proposed Change

This Technical Specification change revises the REACTOR COOLANT SYSTEM Specification 3/4.4.9, PRESSURE/TEMPERATURE LIMITS, by revising the chemistry data in the inset to Figures 3.4-2 and 3.4-3. Specifically, the nickel content for the controlling material plate of the reactor vessel beltline region, A9153-1, is revised from 0.45% to 0.46%. In addition, the proposed amendment revises the BASES 3/4.4.9, PRESSURE/TEMPERATURE LIMITS, to reflect: (i) chemistry data changes (nickel and copper) for reactor vessel beltline materials; (ii) dropweight temperature (T_{NDT}) and Upper Shelf Energy (USE) changes for the reactor vessel circumferential weld 5P6771; (iii) the provision to perform In-Service Leak & Hydrotests (ISLH) using the ISLH Pressure-Temperature Limits whenever fuel is in the reactor vessel; (iv) to require that ISLH tests required by the ASME Code to be completed before the core is made critical and (v) to revise the reference for the Pressure-Temperature Limits requirements from ASME Section III, Appendix G to ASME Section XI, Appendix G.

Basis

This change does not involve a significant hazards consideration for the following reasons:

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   There are no physical changes to any plant equipment created by the proposed changes. The chemistry and material property changes do not impact the ability of the reactor vessel to maintain it's pressure boundary integrity as previously evaluated. The decrease in EOL USE for weld heat 5P6771 is relatively minor and remains above the required value that has been prescribed by the NRC to provide the necessary level of ductility assumed for reactor vessel integrity evaluations. Therefore, the accident initiating and mitigating aspects of the pressure vessel are not affected. In addition, neither the proposed change requiring the ISLH test to be complete before the core is critical nor the proposed change allowing fuel in the reactor vessel during ISLH affects any accident initiating mechanisms. The proposed change requiring the ISLH test to be completed before the core is critical will not increase the consequences of previously evaluated accidents because it conservatively assures the core is subcritical. Although the proposed change allows fuel in the vessel during ISLH utilizing the ISLH Pressure-Temperature (P-T) limits, the consequences of a pressure boundary leak have not changed because ISLH testing is already allowed using the normal plant P-T limits. In addition, the ISLH will be required to be completed before the core is allowed to go critical. The consequences of a leak with fuel in the vessel during ISLH are the same using either the normal P-T limits or the ISLH limits.

Therefore, there would be no increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   There are no physical changes to any plant equipment or new components created by the proposed changes. The chemistry and material property changes do not impact the pressure boundary integrity of the reactor vessel. The decrease in EOL USE for weld heat 5P6771 is relatively minor and remains above the required value that has been prescribed by the NRC to provide the necessary level of ductility assumed for reactor vessel integrity evaluations. Therefore, the accident initiating aspects of the pressure vessel are not affected. In addition, neither the proposed change requiring the ISLH test to be complete before the core is critical nor the proposed change allowing fuel in the reactor vessel during ISLH creates any new accident initiating mechanisms.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in the margin of safety.

   The changes in chemical and material properties do not adversely affect any reactor vessel integrity evaluations, such as PTS or P-T limits. The USE for weld heat 5P6771 does decrease slightly as described in TS Bases Table B 3/4.4-1. However, the predicted EOL USE remains above the value prescribed in 10 CFR 50, Appendix G and is not a significant reduction in the margin of safety. With regard to the proposed changes

allowing fuel in the reactor vessel during ISLH, the existing TS Bases specifically state that fuel is not to be in the reactor vessel when the ISLH P-T curve is utilized. However, this change is consistent with the revised 10 CFR 50, Appendix G rule and as such, is not a significant reduction in the margin of safety.

Therefore, the proposed change does not involve a significant reduction in the margin of safety.

ENCLOSURE 3

SHEARON HARRIS NUCLEAR POWER PLANT
NRC DOCKET NO. 50-400/LICENSE NO. NPF-63
REQUEST FOR LICENSE AMENDMENT
RCS PRESSURE/TEMPERATURE LIMITS

ENVIRONMENTAL CONSIDERATIONS

10 CFR 51.22(c)(9) provides criterion for and identification of licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant hazards consideration; (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite; (3) result in a significant increase in individual or cumulative occupational radiation exposure. Carolina Power & Light Company has reviewed this request and determined that the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of the amendment. The basis for this determination follows:

Proposed Change

This Technical Specification change revises the REACTOR COOLANT SYSTEM Specification 3/4.4.9, PRESSURE/TEMPERATURE LIMITS, by revising the chemistry data in the inset to Figures 3.4-2 and 3.4-3. Specifically, the nickel content for the controlling material plate of the reactor vessel beltline region, A9153-1, is revised from 0.45% to 0.46%. In addition, the proposed amendment revises the BASES 3/4.4.9, PRESSURE/TEMPERATURE LIMITS, to reflect: (i) chemistry data changes (nickel and copper) for reactor vessel beltline materials; (ii) dropweight temperature (T_{NDT}) and Upper Shelf Energy (USE) changes for the reactor vessel circumferential weld 5P6771; (iii) the provision to perform In-Service Leak & Hydrotests (ISLH) using the ISLH Pressure-Temperature Limits whenever fuel is in the reactor vessel; (iv) to require that ISLH tests required by the ASME Code to be completed before the core is made critical and (v) to revise the reference for the Pressure-Temperature Limits requirements from ASME Section III, Appendix G to ASME Section XI, Appendix G.

Basis

The change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) for the following reasons:

1. As demonstrated in Enclosure 2, the proposed amendment does not involve a significant hazards consideration.

2. The proposed amendment does not result in a significant change in the types or increase in the amounts of any effluents that may be released offsite.

   The proposed change does not involve any new equipment or require existing systems to perform a different type of function than they are currently designed to perform. The change does not introduce any new effluents or increase the quantities of existing effluents. As such, the change cannot affect the types or amounts of any effluents that may be released offsite.

3. The proposed amendment does not result in a significant increase in individual or cumulative occupational radiation exposure.

   The proposed change does not result in any physical plant changes or new surveillances which would require additional personnel entry into radiation controlled areas. Therefore, the amendment has no affect on either individual or cumulative occupational radiation exposure.

ENCLOSURE 4
SHEARON HARRIS NUCLEAR POWER PLANT
NRC DOCKET NO. 50-400/LICENSE NO. NPF-63
REQUEST FOR LICENSE AMENDMENT
RCS PRESSURE/TEMPERATURE LIMITS

PAGE CHANGE INSTRUCTIONS

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