Progress Energy

10 CFR 50.90

October 6, 2011
Serial: HNP-11-090

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

Shearon Harris Nuclear Power Plant Unit No. 1
Docket No. 50-400 / Renewed License No. NPF-63

Subject: Request for Additional Information Regarding TAC No. ME5409 (Harris-M5™ cladding)

Reference:
1. Letter from Chris Burton, Progress Energy Carolinas, to USNRC, "Application for Revision to Technical Specification 5.3.1 and Core Operating Limits Report References for M5™ Cladding," HNP-10-124, dated January 13, 2011 (Adams Ascension No. ML110250265)

2. Email from B. L. Mozafari, USNRC, to J. R. Caves, Progress Energy Carolinas, "Request for Additional Information Regarding TAC No. ME5409 (Harris-M5 cladding)," dated September 2, 2011.

Ladies and Gentlemen:

In Reference 1, Carolina Power & Light Company (CP&L), doing business as Progress Energy Carolinas, Inc., requested changes to the Technical Specifications (TS), Appendix A of Renewed Operating License No. NPF-63, for the Shearon Harris Nuclear Power Plant Unit No. 1 (HNP). The proposed changes would modify the HNP TS to permit the use of the AREVA fuel cladding alloy designated as M5™.

In Reference 2, the USNRC issued a request for additional information (RAI). The enclosure to this letter contains HNP's response to that RAI.

CP&L has concluded that the information provided in this response meets the intent of the original submittal (Reference 1) and does not impact the conclusions of the: 1) Technical Analysis, 2) No Significant Hazards Consideration under the standards set forth in 10 CFR 50.92(c), or 3) Environmental Consideration as provided in the original submittal.

This letter contains no new regulatory commitments.

Progress Energy Carolinas, Inc.
Harris Nuclear Plant
P.O. Box 165
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NRR

If there are any questions or if additional information is needed, please contact Dave Corlett at 919-362-3137.

I declare under penalty of perjury that the foregoing is true and correct. Executed on 10-6-11.

Sincerely,

Keith Holbrook
Manager, Support Services
Harris Nuclear Plant

Enclosure: Response to Request for Additional Information Regarding TAC No. ME5409 (Harris-M5™ cladding)

cc: Regional Administrator, USNRC/Region II
Project Manager, Harris Nuclear Plant, USNRC/NRR
Resident Inspector, Harris Nuclear Plant, USNRC
Section Chief, NC Division of Environmental Health

HNP-11-090

Enclosure

Shearon Harris Nuclear Power Plant / Unit No. 1 (HNP)
Docket No. 50-400 / Renewed License No. NPF-63

Application for Revision to Technical Specification 5.3.1 and
Core Operating Limits Report References for M5™ Cladding

Response to Request for Additional Information
Regarding TAC No. ME5409 (Harris-M5™ cladding)

<u>Question 1</u>

The Safety evaluation for Topical Report BAW-10240(P)(A) lists the following four conditions which Framatome (FANP) has accepted:

a) The corrosion limit, as predicted by the best-estimate model will remain below 100 microns for all locations of the fuel.
b) All of the conditions listed in the SEs for all FANP methodologies used for M5™ fuel analysis will continue to be met, except that the use of M5™ cladding in addition to Zircaloy-4 cladding is now approved.
c) All FANP methodologies will be used only within the range for which M5™ data was acceptable and for which the verifications discussed in BAW-l0240(P) or Reference 2 was performed.
d) The burnup limit for this approval is 62 GWd/MTU

Explain in detail, how each of the above conditions has been implemented at HNP Unit 1.

<u>Response 1:</u>

AREVA is responsible for completing safety analysis, neutronic, thermal-hydraulic and fuel mechanical analysis per the NRC-approved methods listed in the HNP Core Operating Limits Report (COLR). BAW-10240 incorporates M5™ material properties into a set of NRC-approved mechanical analysis, small break loss-of-coolant accident (SBLOCA), and non-loss-of-coolant accident methodologies.

a) The restriction that corrosion limit, as predicted by the best-estimate model, will remain below 100 microns for all locations of the fuel is implemented in AREVA design processes. A scoping study of M5™ implementation was performed and confirmed that the fuel mechanical limits can be satisfied for the HNP 17 x 17 HTP fuel design. This is implemented by AREVA, is subject to CP&L owner review and audit, and is subject to NRC audit.

b) AREVA implements the reload specific design. Conditions from approved Safety Evaluations are incorporated as restrictions in AREVA design procedures and guidelines that control the core reload designs provided to Harris Nuclear Plant. This is implemented by AREVA, is subject to CP&L owner review and audit, and is subject to NRC audit.

c) AREVA implements the reload specific design. Limitations to ensure FANP methodologies will be used only within the range for which M5™ data was acceptable and for which the verifications discussed in BAW-l0240(P) or Reference 2 was performed are incorporated as restrictions in AREVA design procedures and guidelines that control the core reload designs provided to Harris Nuclear Plant. This is implemented by AREVA, is subject to CP&L owner review and audit, and is subject to NRC audit.

d) The burnup limitation is not a change for the 17 x 17 HTP fuel assembly. Burnup limits identified in approved methodologies are contained in HNP core functional requirements and AREVA design processes, which are currently limited to 62 GWd/MTU. This is implemented by AREVA, is subject to CP&L owner review and audit, and is subject to NRC audit.

Question 2

The safety evaluation for EMF-2310, Revision 0 topical report (Section 5.0), has restated one of the restrictions (Number 1) from the safety evaluation for ANF-89-151(P)(A) (Section 2.2, TER Conclusions) as stated below:

The stated application of the S-RELAP5 code is for the events listed above in Table 1. There are other computer codes and methodologies employed for evaluation of the events not listed in the table. For each licensing basis event analyzed, the applicant must, as always, justify the methodology used whether by reference to S-RELAP5 or whatever methodology has been used.

Explain how this restriction is implemented for the upcoming cycle reload analyses when the transition to M5™ cladding occurs.

Response 2

Table 1 lists the events analyzed in the HNP Final Safety Analysis Report (FSAR). Table 1 is exclusive of radiological analyses that do not interface with transient analyses (e.g. Fuel Handling Accident, Waste Gas Decay Tank Rupture, etc.). Radiological analysis is performed in accordance with Alternate Source Term (AST) methodology of Regulatory Guide (RG) 1.183 as described in CP&L's submittal on AST. The NRC acceptance of CP&L's implementation of AST is contained in License Amendment 107. Radiological analyses are not impacted by the cladding material.

The Non-LOCA events that rely on methodologies outside of ANP-89-151 and EMF-2310 are Main Steam Line Break (MSLB), Ejection of a Full Length RCCA and Steam Generator Tube Rupture (SGTR). MSLB is analyzed using a separate AREVA methodology, but in the future could be analyzed using EMF-2310. Rod Ejection is analyzed with an event specific methodology (XN-NF-78-44). SGTR is analyzed using Westinghouse methodology (WCAP-10698),

SGTR is analyzed for margin to overfill and offsite dose consequences. From the perspective of SAFDLs the SGTR is bounded by other more severe depressurization events. The change in the cladding material has negligible impact on the analysis of SGTR margin to overfill or offsite dose consequences.

In conclusion, NRC approved methodologies are used for accident analyses and those methods that do not directly interface with BAW-10240 are not impacted by the change in cladding material.

Question 3

RODEX2 fuel deformation and conductivity models were incorporated in S-RELAP5 for transient and accident analyses. Provide details of the methodology in the fuel model to evaluate fuel thermal conductivity as a function of burnup and temperature, considering all of the effects that take in the fuel during the irradiation in the reactor core.

Response 3

The addition of BAW-10240 (P)(A) as a COLR methodology is specific to the incorporation of M5™ material properties and correlations into other approved AREVA methodologies. BAW-10240(P)(A) does not deal with pellet (UO_2) material.

The RAI applies to burnup dependency on the pellet material. This fuel property is the subject of separate NRC approved methodologies, which are not related to the subject license amendment request. The NRC approved methodologies for RODEX2 is collectively listed in HNP Technical Specification Section 6.9.1.6.2.o , Core Operating Limits Report, Mechanical Design Methodologies. No additions or changes to the TS 6.9.1.6.2.o are being made as part of the subject license amendment request. A conservative penalty has been assessed for pellet material thermal conductivity degradation.

Question 4

Provide a list of Shearon Harris Unit 1 FSAR Chapter 15 Non-LOCA events that will be either analyzed or dispositioned for the upcoming fuel cycle when the licensee is planning to use M5™ cladding. Also summarize the methodologies and codes used for the analyses.

Response 4

Table 1 provides the requested information for the upcoming fuel cycle 18. Table 1 does not include radiological events that are listed in HNP FSAR Chapter 15.0; those analyses do not interface with BAW-10240. Table 1 is a projection of the expected safety analysis activities that are currently in progress, but are not yet complete.

As described in ANP-89-151 and EMF-2310, non-LOCA events generally consist of two parts. The first part is a system transient. The second part is an assessment of the impact of cycle specific peaking factors on Minimum Departure from Nucleate Boiling Ratio (MDNBR) and Fuel Centerline Melt (FCM) Specified Acceptable Fuel Design Limits (SAFDLs). Main Steamline Break is a special case in that an additional check is needed to confirm the reactivity conditions at limiting state points.

A number of events are dispositioned based on the characteristics of individual events versus other bounding events. The basis for these dispositions is contained in ANP-89-151 and EMF-2310. The dispositioned events have no associated system transient and do not have SAFDL evaluation on a cycle-by-cycle basis.

Within this framework, only one Non-LOCA system transient is being performed to support Cycle 18. The affected system transient is being updated to recover MDNBR margin and the re-analysis is not related to the change in the cladding material. The impacted FSAR events are Uncontrolled Bank Withdrawal at Power (FSAR 15.4.2) and Withdrawal of a Single Full Length RCCA (FSAR 15.4.3.2). These events share a common system transient; the events differ by the peaking factors applied for the respective RCCA configurations.

The Inadvertent Boron Dilution Event (FSAR 15.4.6) is performed for every cycle. For Cycle 18 the methodology of EMF-2310 will be used. The analysis is performed for almost every reload campaign. The changes in the core reactivity dominate the result and the change in the cladding material has negligible effects compared to differences in cycle design.

SBLOCA and LBLOCA are both being revised for Cycle 18. The applicable SBLOCA methodology (EMF-2328) was previously approved for the HNP docket. The LBLOCA is being re-analyzed using AREVA's EMF-2103 methodology; an application to add this COLR methodology is currently pending NRC review and approval.

The neutronics methods used to assess the power distribution and reactivity inputs for the MDNBR and FCM analysis and the thermal hydraulic methods used to assess compliance with SAFDLs are listed in Table 2.

Table 1

FSAR Section	Event Name (ANS category)	Listed in EMF-2310 TER Table 1?	Cycle 18 activity	Cycle 18 Methodology	Comment
15.1.1	FW System Malfunction that Results in a Decrease in FW temperature (II)	No (See Comment)	Event Bounded by FSAR 15.1.3	ANP-89-151	The event is included in the TER table for ANP-89-151 (P)(A) . By the nature of the event it is considered in scope for EMF-2310, even though not listed in the EMF-2310 TER table, due to the applicability of the EMF-2310 method to "Increase in Heat Removal by the Secondary System" event grouping.
15.1.2	FW System Malfunction that Results in an Increase in FW flow (II)	Yes	Evaluate MDNBR and FCM SAFDL	ANP-89-151	System transient to be dispositioned for M5.
15.1.3	Excessive Increase in Secondary Steam Flow (II)	Yes	Evaluate MDNBR and FCM SAFDL	ANP-89-151	System transient to be dispositioned for M5.
15.1.4	Inadvertent Opening of SG relief or Safety Valve (II)	Yes	Event Bounded by FSAR 15.1.3 (at power)	ANP-89-151	Event bounded by FSAR 15.1.5 with no fuel failure after reactor trip
15.1.5	Steam System Piping Failures (IV)	Yes	Evaluate MDNBR and FCM SAFDL and reactivity interface	EMF-84-093	System transient not currently performed using EMF-2310. System transient to be dispositioned for M5.

Table 1 (cont.)

FSAR Section	Event Name (ANS category)	Listed in EMF-2310 TER Table 1?	Cycle 18 activity	Cycle 18 Methodology	Comment
15.2.1	BWR event NA to HNP	N/A	-	-	-
15.2.2	Loss of External Electrical Load (II)	Yes	Event Bounded by FSAR 15.2.3	ANP-89-151	-
15.2.3	Turbine Trip (II)	Yes	Evaluate MDNBR SAFDL	ANP-89-151	System transient to be dispositioned for M5.
15.2.4	Inadvertent closure of Main Steam Isolation Valves (II)	Yes	Event Bounded by FSAR 15.2.3	ANP-89-151	-
15.2.5	Loss of Condenser Vacuum (II)	Yes	Event Bounded by FSAR 15.2.3	ANP-89-151	-
15.2.6	Loss of Non-Emergency AC Power to the Station Auxiliaries (II)	Yes	Event Bounded by FSAR 15.3.2	ANP-89-151	-
15.2.7	Loss of Normal Feedwater Flow (II)	Yes	No SAFDL analysis required.	ANP-89-151	System transient to be dispositioned for M5.
15.2.8	Feedwater System Pipe Break (IV)	Yes	No SAFDL analysis required.	ANP-89-151	System transient to be dispositioned for M5.

Table 1 (cont.)

FSAR Section	Event Name (ANS category)	Listed in EMF-2310 TER Table 1?	Cycle 18 activity	Cycle 18 Methodology	Comment
15.3.1	Partial Loss of Forced Reactor Coolant Flow (II)	Yes	Event Bounded by FSAR 15.3.2 with more restrictive ANS II acceptance criteria	ANP-89-151	-
15.3.2	Complete Loss of Forced Reactor Coolant Flow (III)	Yes	Evaluate to ANS II MDNBR SAFDL	ANP-89-151	System transient to be dispositioned for M5.
15.3.3	Reactor Coolant Pump Shaft Seizure (Locked Rotor) (IV)	Yes	Evaluate number of fuel assemblies that exceed DNB criteria	ANP-89-151	System transient to be dispositioned for M5.
15.3.4	Reactor Coolant Pump Shaft Break (IV)	Yes	Event Bounded by FSAR 15.3.3	ANP-89-151	-
15.4.1	Uncontrolled Rod Cluster control Assembly Bank Withdrawal from a Subcritical or Low Power Startup Condition (II)	Yes	Evaluate MDNBR and FCM SAFDL	ANP-89-151	System transient to be dispositioned for M5.

Table 1 (cont.)

FSAR Section	Event Name (ANS category)	Listed in EMF-2310 TER Table 1?	Cycle 18 activity	Cycle 18 Methodology	Comment
15.4.2	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (II)	Yes	Evaluate MDNBR and FCM SAFDL	EMF-2310	New system transient to be performed for Cycle 18.
15.4.3.1	Dropped Full Length RCCA or RCCA Bank (II)	Yes	Evaluate MDNBR and FCM SAFDL	ANP-89-151	System transient to be dispositioned for M5.
15.4.3.2	Withdrawal of a Single Full Length RCCA (III)	Yes	Evaluate number of fuel assemblies that exceed DNB criteria and FCM	EMF-2310	New system transient is performed for Cycle 18.
15.4.3.3	Statically Misaligned RCCA or Bank (II)	Yes	Evaluate MDNBR and FCM SAFDL	ANP-89-151	No transient analysis performed. M5™ properties included in neutronic input.
15.4.4	Startup of Inactive Reactor Coolant Pump at Incorrect Temperature	Yes	Event bounded by FSAR 15.4.1 in Modes 3-5. Event precluded by administrative control in Modes 1 and 2.	ANP-89-151	-
15.4.5	BWR event NA to HNP	N/A	-	-	-

Table 1 (cont.)

FSAR Section	Event Name (ANS category)	Listed in EMF-2310 TER Table 1?	Cycle 18 activity	Cycle 18 Methodology	Comment
15.4.6	Chemical and Volume Control System Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant (II)	Yes	Mode 1 bounded by FSAR 15.4.2. Other modes covered by cycle specific reactivity calculations	EMF-2310	No transient analysis performed. M5™ properties included in reactivity calculations
15.4.7	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position (III)	Yes	Evaluate number of fuel assemblies that exceed DNB and FCM criteria	ANP-89-151	No transient analysis performed. M5™ properties included in neutronic input.
15.4.8	Spectrum of Rod Cluster Control Assembly Ejection Accidents (IV)	Yes	Evaluate number of fuel assemblies that exceed DNB and FCM criteria	XN-NF-78-44 and ANP-89-151	System transient to be dispositioned for M5.
15.4.9	BWR event NA to HNP	N/A	-	-	-

Table 1 (cont.)

FSAR Section	Event Name (ANS category)	Listed in EMF-2310 TER Table 1?	Cycle 18 activity	Cycle 18 Methodology	Comment
15.5.1	Inadvertent Operation of the Emergency Core Coolant System During Power Operation (II)	Yes	No impact on SAFDL as reactor trip occurs at event start.	-	-
15.5.2	CVCS failures that increase RCS inventory	Yes	Event bounded by FSAR 15.5.1 and 15.4.6	-	-
15.6.1	Inadvertent Opening of Pressurizer Safety or Power Operated Relief Valve (II)	Yes	Evaluate MDNBR SAFDL	ANP-89-151	System transient to be dispositioned for M5.
15.6.2	Break in Instrument Line or Other Line From RCPB that penetrates Containment	Yes (see comment)	-	-	Radiological analysis uses bounding non-mechanistic release of RCS. Analysis does not interface with BAW-10240 referenced methods.

Table 1 (cont.)

FSAR Section	Event Name (ANS category)	Listed in EMF-2310 TER Table 1?	Cycle 18 activity	Cycle 18 Methodology	Comment
15.6.3	SGTR	See comment	Disposition	WCAP-10698	The methodology for SGTR overfill and SGTR offsite dose are not performed using AREVA methodology. The methodology employed was approved by NRC in the HNP License Amendment No. 107.
15.6.3	Steam Generator Tube Rupture (IV) Margin to Overfill (MTO)	See comment for line above.		-	-
15.6.4	BWR event NA to HNP	N/A	-	-	-

Table 2

Technical Specification entry number	Topical Report Number	Title	Application
6.9.1.6.2.a	XN-75-27 (P)(A)	Exxon Nuclear Neutronics Design Methods for Pressurized Water Reactors	The low power physics testing methodology for "rod swap" is contained in this topical report. The method is used to confirm RCCA reactivity measurements. This method is not used for core design.
6.9.1.6.2.b	ANF-89-151(P)(A)	ANF-RELAP Methodology for Pressurized Water Reactors Analysis of Non-LOCA Chapter 15 Events	Non-LOCA methodology
6.9.1.6.2.c	XN-NF-82-21	Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations	This method provides mixed core penalties for MDNBR analyses. The switch to M5™ is not considered a "mixed core" for the purposes of thermal hydraulic analyses since the flow channel geometry of the M5™ and Zircaloy 4 assemblies is the same.
6.9.1.6.2.d	XN-75-32(P)(A)	Computational Procedure of Evaluation Fuel Rod Bowing	Method used to determination when additional MDNBR penalty applied.
6.9.1.6.2.e	EMF-84-093	Steam Line Break Methodology for PWR	MSLB methodology
6.9.1.6.2.f	EMF-2087	SEM/PWR-98: ECCS Evaluation Model for PWR LBLOCA Applications	For cycle 18, EMF-2087 methodology being replaced with EMF-2103 based methodology in separate license amendment.

Table 2 (cont.)

Technical Specification entry number	Topical Report Number	Title	Application
6.9.1.6.2.g	XN-NF-78-44	A Generic analysis of the control Rod Ejection Transient for PWR	Methodology employs ANF-RELAP
6.9.1.6.2.h	ANP-88-054 (P)(A)	PDC-3: Advanced Nuclear Fuels Corporation Power Distribution Control for Pressurized Water Reactors	Provides neutronics inputs to safety analyses.
6.9.1.6.2.i	EMF-92-081 (P)(A)	Statistical Setpoint /Transient methodology for Westinghouse Type Reactors	Method for statistical application of DNB correlation and checks of OPΔT and OTΔT trip setpoints
6.9.1.6.2.j	EMF-92-153 (P)(A)	HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel	DNB correlation for evaluation of SAFDL
6.9.1.6.2.k	XN-NF-82-49 (P)(A)	Exxon Nuclear Company Evaluation Model EXEM PWR Small Break Model	Not used with M5™ fuel.
6.9.1.6.2.l	EMF-96-029(P)(A)	Reactor Analysis System for PWRs	Provides neutronic input to safety analyses.
6.9.1.6.2.m	EMF-2328 (P)(A)	PWR Small Break LOCA Evaluation Model	SBLOCA methodology compatible with M5™ cladding.
6.9.1.6.2.n	EMF-2310	SRP Chapter 15 Non-LCOA Methodology for Pressurized Water Reactors	Non-LOCA methodology.