Lewis Sumner
Vice President
Hatch Project Support

Southern Nuclear Operating Company, Inc.
40 Inverness Parkway
Post Office Box 1295
Birmingham, Alabama 35201

Tel 205.992.7279
Fax 205.992.0341

SOUTHERN COMPANY
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Docket No. 50-321

HL-6175

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Edwin I. Hatch Nuclear Plant - Unit 1
Response to Request for Additional Information on
Technical Specification Revision Request:
Deferral of Type A Containment Integrated Leak Rate Test (ILRT)

Ladies and Gentlemen:

In response to a request from the NRC Project Manager for Edwin I. Hatch Nuclear Plant on January 22, 2002, Enclosures 1 and 2 provide responses to requests for additional information which were initially responded to electronically on December 27, 2001, and January 7, 2002, respectively.

Mr. H. L. Sumner, Jr. states he is Vice President of Southern Nuclear Operating Company and is authorized to execute this oath on behalf of Southern Nuclear Operating Company, and to the best of his knowledge and belief, the facts set forth in this letter are true.

Respectfully submitted,

Lewis Sumner

H. L. Sumner, Jr.

Sworn to and subscribed before me this 24th *day of* January 2002.

Elaine E. Bolton
Notary Public

Commission Expiration Date: 5-25-2003

IFL/eb

Enclosures:
1. Response to Request of December 27, 2001
2. Response to Request of January 7, 2002

cc: (See next page.)

A001

cc: Southern Nuclear Operating Company
Mr. P. H. Wells, Nuclear Plant General Manager
SNC Document Management (R-Type A02.001)

U.S. Nuclear Regulatory Commission, Washington, D.C.
Mr. L. N. Olshan, Project Manager - Hatch

U.S. Nuclear Regulatory Commission, Region II
Mr. L. A. Reyes, Regional Administrator
Mr. J. T. Munday, Senior Resident Inspector – Hatch

Enclosure 1

Edwin I. Hatch Nuclear Plant - Unit 1
Response to Request for Additional Information on
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Deferral of Type A Containment Integrated Leak Rate Test (ILRT)

Response to Request of December 27, 2001

NRC Request No. 1:

Has HNP submitted any relief requests concerning gaskets, seals, and bolted connections?

SNC Response:

Yes.

Relief Request RR-MC-1 (Seals (including O-rings) and gaskets of Class MC pressure retaining components, Examination Category E-D, Item Numbers E5.10 and E5.20). Approved in SE dated February 11, 2000.

Relief Request RR-MC-6 (Class MC pressure retaining bolting requiring visual examination (VT-1) per Category E-G, Item E8.10, in accordance with Subarticle IWE-3515.1). Approved in SE dated February 11, 2000.

Relief Request RR-MC-8 (Class MC pressure retaining bolting for bolted connections that have not been disassembled and reassembled during the inspection interval). Approved in SE dated October 4, 2000.

NRC Request No. 2:

What inspections are performed on gaskets, seals, and bolted connections?

SNC Response:

Per RR-MC-1, Leak-tightness of the seals (including O-rings) and gaskets will be confirmed in accordance with 10 CFR 50, Appendix J. If a seal (including O-rings) or gasket is replaced, it will be visually inspected by maintenance personnel before re-assembly or closure. Also, an as-left Appendix J leakage test will be performed after installation to ensure leak-tightness.

Per RR-MC-6, Bolting material will be examined in accordance with the inservice standards of the 1992 Edition, with 1992 Addenda of ASME Section XI, Subarticle IWB-3517.1 Standards for Examination Category B-G-1, Pressure Retaining Bolting Greater Than 2 in. in Diameter, and Examination Category B-G-2, Pressure Retaining Bolting 2 in. and Less in Diameter.

Per RR-MC-8, ASME Code Case N-604 will be used for alternate examination of pressure retaining bolting in lieu of torque or tension testing.

NRC Request No. 3:

What App. J test is performed on the stainless steel bellows? Type A or B?

SNC Response:

All bellows are Type B tested in accordance with 10 CFR 50, Appendix J, Option B. All bellows are also tested during the Appendix J, Type A test.

NRC Request No. 4:

How are inaccessible areas handled?

SNC Response:

IWE-1232 provides criteria applicable to inaccessible surface areas. IWE-1232(a) states; "Portions of Class MC containment vessels, parts, and appurtenances that are embedded in concrete or otherwise made inaccessible during construction of the vessel or as a result of vessel repair, modification, or replacement are exempted from examination, provided:

1. No openings or penetrations are embedded in the concrete;

2. All welded joints that are inaccessible for examination are double butt-welded and are fully radiographed and, prior to being covered, are tested for leak tightness using a gas medium test, such as halide Leak Detector Test;

3. All welded joints that are not double butt welded remain accessible for examination from the weld side; and

4. The vessel is leak rate tested after completion of construction, repair, or replacement to the leak rate requirements of the Design Specification."

<u>Drywell Shell</u>

The exterior surface of the drywell shell, with the exception of the drywell head, is exempt from examination per IWE-1220(b) and IWE-1232(a). The exterior surface is inaccessible due to the concrete shield wall and the 2" air gap. All shell welds were solution film tested at 56 psig after completion of construction which met the construction code requirements and IWE-1232(a)(2) for Unit 2.

The interior and exterior embedded portions of the drywell shell (i.e., below the 114 foot elevation) are exempt from examination per IWE-1220(b) and IWE-1232(a). Shell welds below the 114'-8" elevation were examined per the ASME Section III Code, Subsection NE (1971 Edition including the Summer 1971 Addenda, FSAR 3.2.2, i.e., halide leak tested), and all shell welds were solution film tested at 56 psig after completion of construction which met the construction code requirements and thus the requirements of IWE-1232(a)(2).

The functionality of the drywell air gap and sand cushion drain lines has been confirmed for both Hatch Units. Visual examination of the drains was conducted utilizing a video probe to insure that the drains would function in the event of a refueling bellows leak. In addition to IWE required examinations, each drain line is visually examined each outage while the reactor refueling cavity is flooded to ensure there is no evidence of moisture or leakage. These examinations insure that there is no leakage which would lead to degradation of the exterior of the drywell shell. Additionally, sample ultrasonic thickness measurements of each drywell shell course were taken and compared to the fabrication requirements. In all instances, the actual shell thickness was = or > the required nominal thickness.

Suppression Pool Interior Surfaces

Virtually 100% of the interior non-submerged suppression pool surfaces (vapor space) are accessible for visual examination from the 114'-0" interior catwalk, or from the top of the vent header if the provisions of 10CFR50.55a(b)(2)(x)(B) are applied. Visual examination from these vantage points should provide adequate access to perform visual examinations of these surfaces.

Visual examination of this suppression pool vapor space is performed every outage by site QC personnel. Additionally, a contractor performed visual examination of the vapor space beginning in the spring 1991 refueling outage. This contractor performed subsequent examinations during the 1995 and 1997 refueling outages. The results of these examinations indicate only minor coating loss, and virtually no degradation of the shell material.

Suppression Pool Interior Submerged Surfaces

The submerged surfaces of the suppression pool are only accessible for visual examination using underwater divers or by draining the pool. Plant Hatch began an extensive desludging, visual examination, and patch coating repair program during the spring 1991 refueling outage. The results of these activities indicated that the Unit 2 submerged surfaces were in good condition and examinations were scheduled again for the 1995 outage. Visual examination, desludging, and patch coating repairs have been implemented during each outage since 1995 and a long range suppression pool maintenance program is in place.

The results of these examinations indicated that there was some coating degradation and shell pit measurements were taken to determine the maximum depth and the rate of degradation. Review and evaluation of the latest examination results reveal a maximum pit depth of approximately 0.040" with an average corrosion rate of 1.74 mils (0.00174") per year. The nominal thickness of the submerged area is 0.594" with a minimum design thickness of 0.440". Sample ultrasonic thickness measurements were performed in 1998 which confirmed the shell thickness to be greater than design requirements. Plant Hatch plans to continue desludging, examination, and spot coating repair.

No other areas of the containment are considered possibly inaccessible as related to ASME Section XI, Class MC.

Enclosure 2

Edwin I. Hatch Nuclear Plant - Unit 1
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Response to Request of January 7, 2002

NRC Request:

Inspections of some reinforced concrete and steel containment structures have found degradation on the uninspectable (embedded) side of the drywell steel shell and steel liner of the primary containment. These degradations cannot be found by visual (i.e., VT-1 or VT-3) examinations unless they are through the thickness of the shell or liner, or, 100% of the uninspectable surfaces are periodically examined by ultrasonic testing. Please provide information addressing how potential leakage under high pressure during core damage accidents is factored into the risk assessment related to the extension of the ILRT.

SNC Response:

ILRT EXTENSION IMPACT ON CONTAINMENT DEGRADATION SITES

All penetrations, attachments, hatches, bolting surfaces, and structural members (including the steel shell) are considered in the PRA structural analysis to assess the failure pressure and failure location of containment. The probability of the degradation of these boundaries leading to containment failure is included in the Hatch probabilistic risk assessment and in the ILRT extension evaluation. Specifically, the intact containment case, EPRI Containment Failure Class 1, includes a leakage term that reflects the potential for small leakage at penetrations or the drywell shell independent of the ILRT interval. Further, the Containment Failure Classes 3a and 3b model the potential for larger leakage impacts including the ILRT interval extension. Classes 3a and 3b include the potential that the leakage is due to a containment shell failure, bellows failure, or penetration failure and some fraction of these are affected by the ILRT interval extension. A model based on NUREG 1493 results is used to predict the likelihood of having a small/large breach that is undetected by the Type A ILRT test. These events are represented by the "Class 3" sequence depicted in EPRI TR-104285. The two failure modes (Class 3a and 3b) were considered to ensure proper representation of available data. These two failure modes are Class 3a (small breach) and Class 3b (large breach). Changes in the calculated frequencies of Classes 3a and 3b reflect the increased frequencies of these failure modes associated with extending the ILRT interval.

The assessment shows that even with the increased potential to have an undetected containment flaw or leak path due to extending the ILRT interval, the increase in risk is insignificant.