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SUBJECT: Forwards Rev 1 of third 10-yr ISI interval Requests for Relief ONS-001 through ONS-006.Rev 1 had previously been omitted at upper right margin of each request.No changes made in body of requests since last submittal dtd 940614.

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DUKE POWER

September 15, 1994

U.S. Nuclear Regulatory Commission
Attention Document Control Desk
Washington, DC 20555

Subject: Duke Power Company
Oconee Nuclear Station, Units 1,2, and 3
Docket No. 50-269, 50-270, 50-287
Third Ten Year Inservice Inspection Interval
Request for Reliefs ONS 001-006 Rev 1

Requests for Relief ONS-001 through ONS-006 were previously submitted to the NRC per J. W. Hampton's letter dated 14 June 1994. Revision 1 of these Requests for Relief are enclosed for re-submittal. No changes in the body of the requests have been made since the last submittal to the NRC. "Rev. 1" had previously been omitted at the upper right margin of each request.

If there are any questions or further information is needed you may contact D. A. Nix at (803) 885-3634.

Very truly yours,

J.W. Hampton

J. W. Hampton
Site Vice President

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A047
/1

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2600 Bull St.
Columbia, SC 29201

DUKE POWER COMPANY

Request for Relief From
Inservice Inspection Requirement

Station: **Oconee**

Unit: **1,2 & 3**

Requesting Department: Nuclear Generation

Reference Code: ASME Boiler and Pressure Vessel Code, Section XI 1989 Edition, no Addenda

I. Component for which exemption is requested:

a. Name and Identification Number:

Core Flood Nozzle-to-Safe End and Safe End-to-Pipe Welds Drawing OM-201-92 (Attachment "A"), OM-1201-1528 (Attachment "B") and B&W Drawing 149906E (Attachment "C").

Oconee 1

Item No.	ID No.	Description
B09.011.090A	1-53A-02-43L	SE to Pipe Weld
B09.011.100A	1-53A-01-1L	SE to Pipe Weld
B05.010.001	1-RPV-WR53	Noz to SE Weld
B05.010.002	1RPV-WR53A	Noz to SE Weld

Oconee 2

Item No.	ID No.	Description
B09.011.011A	2-53A-8-63	SE to Pipe Weld
B09.011.013A	2-53A-8-64	SE to Pipe Weld
B05.010.001	2-RPV-WR53	Noz to Pipe Weld
B05.010.002	2-RPV-WR53A	Noz to Pipe Weld

Oconee 3

Item No.	ID No.	Description
B05.010.001	3-RPV-WR53	Noz to SE Weld
B05.010.002	3RPV-WR53A	Noz to SE Weld
B09.011.040A	3-53A -15-44	SE to Pipe Weld
B09.011.045A	3-53A-16-01	SE to Pipe Weld

b. Function:

Provides reactor vessel core flooding capability

c. ASME Section XI Code Class:

Class 1

d. Construction Code and Class (If Applicable):

ASME Section III, Class 1

e. Valve Category (If Applicable):

NA

II. Reference Code Requirement that has been determined to be impractical:

Table IWB-2500-1; Category B-F; Item No.B05.10 - Surface Examination and Category B-J; Item B09.011 - Surface Examination

III. Basis for Requesting Relief:

Relief requested from surface examination only.

Approximately 40 man-hours would be required to prepare each of the two core flood nozzle safe ends for inspection. The preparation would involve removal of the refueling canal seal plate, shielding bricks, shielding supports in the nozzle area, and insulation. The radiation levels in this area are expected to be 0.51 R/hr. An alternative approach is to enter from the bottom of the vessel and build a scaffold approximately 30 feet high to reach the nozzles. This approach would require approximately 80 man-hours, 40 in 0.51 R/hr. area and the other 40 in the 1-2 R/hr. radiation present at the bottom of the reactor vessel, for a total exposure of 80-140 Man-Rem. Shielding is considered impractical in this area. Any remote inspection would require practically the same preparation work.

IV. Alternate Examination:

Welds will be examined by automated UT from the inside surface using the technique demonstrated by the B&W Owners Group in Lynchburg VA, on August 11, 1993 for the NRC. This will provide an acceptable level of quality and safety and not endanger the public health and safety.

V. Implementation Schedule:

Core Flood nozzle-to-safe end and safe end-to-pipe welds will be inspected at or near the end of the third ten year interval.

Evaluated By: A. J. Hogge Jr Date 8-18-94

Reviewed By: J Barlow Date 8/23/94

DUKE POWER COMPANY

Request for Relief From
Inservice Inspection Requirement

Station: **Oconee**

Unit: **1,2 & 3**

Requesting Department: Nuclear Generation

Reference Code: ASME Boiler and Pressure Vessel Code, Section XI 1989 Edition, no Addenda

I. Component for which exemption is requested:

a. Name and Identification Number:

Reactor Vessel Nozzle to Pipe Welds.

Isometrics: System 50, ISO 26 (Unit 1) (Attachment "A")
System 50, ISO 9 (Unit 2) (Attachment "B")
System 50, ISO 29 (Unit 3) (Attachment "C")

Oconee 1

Item No.	ID No.
B09.011.065A	1-PHA-1
B09.011.015A	1-PDA1-9
B09.011.031A	1-PDA2-9
B09.011.077A	1-PHB-1
B09.011.047A	1-PDB1-9
B09.011.063A	1-PDB2-9

Oconee 2

Item No.	ID No.
B09.011.019A	2-PHA-1
B09.011.032A	2-PDA1-8
B09.011.033A	2-PDA2-8
B09.011.021A	2-PHB-1
B09.011.034A	2-PDB1-8
B09.011.035A	2-PDB2-8

Oconee 3

Item No.	ID No.
B09.011.001A	3-PHA-1
B09.011.018A	3-PDA1-8
B09.011.020A	3-PDA2-8
B09.011.003A	3-PHB-1
B09.011.022A	3-PDB1-8
B09.011.024A	3-PDB2-8

b. Function:

Provides reactor coolant flow to steam generators

c. ASME Section XI Code Class:

Class 1

d. Construction Code and Class (If Applicable):

ASME Section III, Class 1

e. Valve Category (If Applicable):

NA

II. Reference Code Requirement that has been determined to be impractical:

Table IWB-2500-1; Category B-J; Item B9.11; Surface examination

III. Basis for Requesting Relief:

Relief requested from surface examination only.

There are four inlet and two outlet nozzle to pipe welds in each Oconee Reactor Coolant System. These nozzles are SA 508 Cl. 2, welded to A106 Gr. C pipe. The inlet nozzle welds are 33.50" in diameter, 2.33" nominal wall thickness and the outlet nozzles are 36" diameter, 2.86 "nominal wall thickness. These welds will be volumetrically inspected from the inside surface using a contact automated ultrasonic technique, which will not require access to the OD surface of the weld. Preparing these welds for surface inspection will require removal of the refueling canal seal plate, shielding bricks, shielding supports in the nozzle areas, and insulation. This would require approximately 300 man-hours of work in a 700-1000 MR/hour area. Shielding would be impractical in this area due to the limited space and close proximity to the reactor vessel.

IV. Alternate Examination:

Welds will be examined by automated UT from the inside surface using the technique demonstrated by the B&W Owners Group in Lynchburg VA, on August 11, 1993 for the NRC. This will provide an acceptable level of quality and safety and not endanger the public health and safety.

V. Implementation Schedule:

The outlet nozzle to pipe welds were examined during the third inspection period of the second interval. The inlet nozzle to pipe welds will be examined during the third inspection period of the third interval.

Evaluated By: A. J. Hogge Jr Date 8-18-94

Reviewed By: [illegible] Date 8/23/94

DUKE POWER COMPANY
Request for Relief From Inservice Inspection Requirement

Station: **Oconee**

Unit: **1,2 & 3**

Requesting Department: Nuclear Generation

Reference Code: 1989 Edition of ASME Section XI, no addenda

I. Component for which exemption is requested

a. Name and Identification Number:

NA

b. Function:

NA

c. ASME Section XI Code Class:

1, 2 and 3

d. Construction Code and Class (If Applicable):

NA

e. Valve Category (If Applicable):

NA

II. Reference Code Requirement that has been determined to be impractical:

IWA-5250 CORRECTIVE MEASURES

Paragraph IWA-5250 (a)(2) "if leakage occurs at a bolted connection, the bolting shall be removed, VT-3 visually examined for corrosion, and evaluated in accordance with IWA-3100 ".

III. Basis for Requesting Relief:

Removal of all bolting at a bolted connection is not always required to assure the connection has not degraded. In addition, some connections are more difficult to seal after disassembly when compared to retorquing an already assembled connection. Complete disassembly of a connection in a radiation area would also increase personnel exposure. ASME has recognized such situations and changed this requirement to allow removal and evaluation of one bolt closest to the leak in a bolted connection identified as leaking.

IV. Alternate Examination:

In lieu of removing all bolting at leaking connections, we intend to adopt the requirements of the 1990 addenda that allows for only the one bolt closest to the leak be removed and examined visually by a VT-3 inspector.

This alternative examination will provide an acceptable level of quality and safety and not endanger the public health and safety.

V. Implementation Schedule:

We intend to implement this alternate examination with the start of refueling outage number 16 (first outage of the third interval for Unit 1) ; Outage Number 15 (first outage of the third interval for Unit 2); Outage Number 15 (first outage of third interval for Unit 3), and thereafter for the entire third interval.

Evaluated By: A. J. Hogge Jr Date 8-18-94

Reviewed By: J Barlow Date 8/23/94

DUKE POWER COMPANY

Request for Relief From
Inservice Inspection Requirement

Station: **Oconee**

Unit: **1,2 & 3**

Requesting Department: Nuclear Generation

Reference Code: ASME Section XI, 1989 Edition, no addenda

I. Component for which exemption is requested:

a. Name and Identification Number:

Control Rod Drive Mechanism (CRDM) Motor Tube To Nozzle Pressure Retaining Bolting. Diamond Power Dwg. 7032551058-D (Attachment "A")

Oconee 1

Item No.	ID No.
B07.080.001	1-RPV-CRD-BOLTS

Oconee 2

Item No.	ID No.
B07.080.001	2-RPV-CRD-BOLTS

Oconee 3

Item No.	ID No.
B07.080.001	3-RPV-CRD-BOLTS

b. Function:

To secure the motor tube to the reactor vessel and to seal off the RCS to prevent leakage

c. ASME Section XI Code Class:

Class 1

d. Construction Code and Class (If Applicable):

N/A

e. Valve Category (If Applicable):

N/A

II. Reference Code Requirement that has been determined to be impractical:
ASME Boiler and Pressure Vessel Code Section XI, 1989 Edition, no addenda, paragraph IWB-2430(a), which requires that bolting showing indications exceeding the standards allowed by IWB-3000 require additional number of components be examined. The purpose of this Code requirement is to assure other CRDM bolting material is not degraded due to the same cause.

III. Basis for Requesting Relief:

During each refueling outage since December 1980, all CRDMs have been visually examined for evidence of RCS leakage. This inspection is documented as part of the Oconee response to Generic Letter 88-05 and IE Bulletin 82-02. During the Unit 3 End of Cycle 11 Refueling Outage, 11 CRDMs exhibited evidence of RCS leakage (see attached figure). Each of these CRDMs were disassembled and examined pursuant to the requirements of Table IWB-2500-1 Item B7.80. This process involved approximately 24 person-rem of exposure.

Of the 11 disassembled CRDMs only 1/2 of 1 nut ring exceeded the criteria of IWB-3000. Damage to this nut ring was specifically due to corrosion as a result of exposure to RCS leakage. As a result of this indication, IWB-2430(a) requires an additional sample of CRDMs be examined.

The requirements of IWB-2430(a) have been determined to be impractical, because it may unnecessarily require disassembly and VT-1 examination of CRDM bolting material which was not affected by RCS leakage. Further, disassembly and VT-1 examination of the additional CRDMs will involve approximately 20 - 30 person-rem of radiation exposure to personnel.

IV. Alternate Examination:

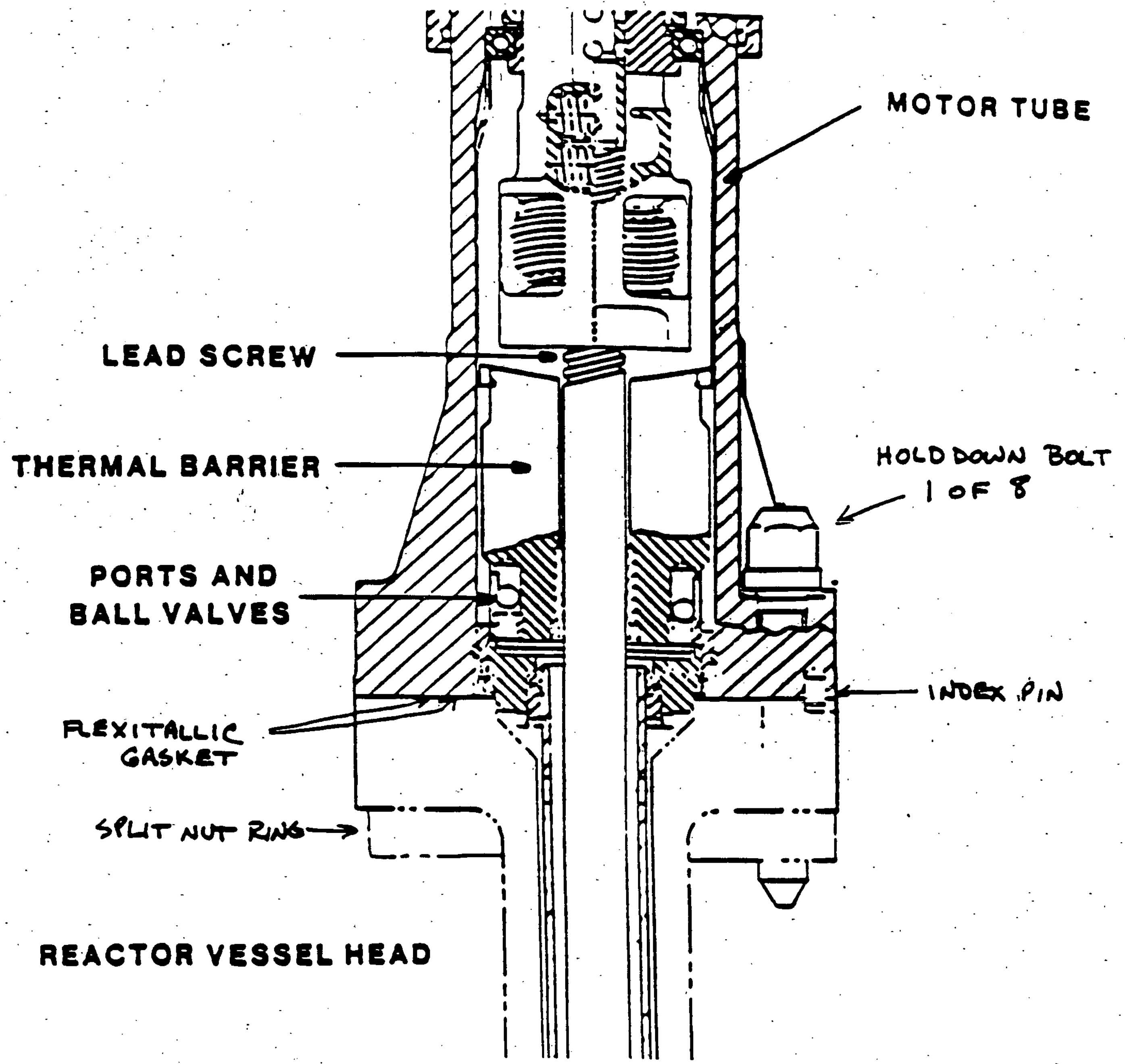
During each refueling outage all CRDM flanges will be visually examined per station procedures for evidence of leakage in compliance with the Oconee Nuclear Station response to the NRC Generic Letter 85-005 and IE Bulletin 82-02. Corrective action will be based upon the results of those examinations, and will include replacement of all affected bolting. Inspection of any required additional samples of bolting material during CRDM maintenance not associated with flange leakage will be performed in accordance with the requirements of IWB-2430(a). This will provide an acceptable level of quality and safety and not endanger the public health and safety.

V. Implementation Schedule:

This request was approved for the second ten year inservice inspection interval and will be continued for the third ten year inservice inspection interval, for Units 1,2 & 3.

Evaluated By: A. J. Hogge, Jr Date 8-18-94

Reviewed By: J. Barlow Date 8/23/94



TITLE	NOTES	
CONTROL ROD DRIVE MECHANISM	THERMAL BARRIER	OC-PNS-CRD-10, DATE 12-3-86
		Diamond Power 7032551058-D
		DMC / ARB

DUKE POWER COMPANY

Request for Relief From
Inservice Inspection Requirement

Station: **Oconee**

Unit: **1, 2 & 3**

Requesting Department: Nuclear Generation

Reference Code: ASME Section XI, 1989 Edition, no addenda

I. Component for which exemption is requested:

a. Name and Identification Number:

Control Rod Drive Mechanism (CRDM) Motor Tube To Nozzle Pressure Retaining Bolting. Diamond Power Dwg. 7032551058-D (Attachment "A")

Oconee 1

Item No.	ID No.
B07.080.001	1-RPV-CRD-BOLTS

Oconee 2

Item No.	ID No.
B07.080.001	2-RPV-CRD-BOLTS

Oconee 3

Item No.	ID No.
B07.080.001	3-RPV-CRD-BOLTS
B07.080.002	3-RPV-CRD-RINGS

b. Function:

To secure the motor tube to the reactor vessel and to seal off the RCS to prevent leakage.

c. ASME Section XI Code Class:
Class 1

d. Construction Code and Class (If Applicable):

N/A

e. Valve Category (If Applicable):

N/A

II. Reference Code Requirement that has been determined to be impractical:

ASME Boiler and Pressure Vessel Code Section XI, 1989 Edition, no addenda paragraph IWB-2500 Item Number B07.80 requires CRDM bolting material to undergo VT-1 visual examination when disassembled. The intent of this code requirement is to assure the disassembled bolting material is acceptable for re-use and to increase confidence that there is not a generic problem occurring that should be further investigated through additional inspections.

III. Basis for Requesting Relief:

Per Oconee Nuclear Station policy, CRDM bolting material removed due to exposure to RCS leakage is not re-used because the excessive boron deposit degradation makes it unsuitable for further use. It is replaced during maintenance for flange leakage by new material that has a pre-service examination performed on it prior to installation. The boron deposit degradation makes it virtually impossible to perform a meaningful inservice inspection. As a result, Table IWB-2500-1 Item Number B7.80 requirements for VT-1 examination of bolting material when CRDMs are disassembled due to RCS leakage indications (see attachment " A") are unnecessary since the material will not be re-used and is in no condition to disclose any possible generic problems. In addition, VT-1 examination of the bolting material which will not be re-used involves significant unnecessary radiation exposure to personnel.

IV. Alternate Examination:

During each refueling outage all CRDM flanges will be visually examined per station procedures for evidence of leakage in compliance with the Oconee Nuclear Station response to the NRC Generic Letter 85-005 and IE Bulletin 82-02. Corrective action (including replacement of affected bolting) will be based upon the results of those examinations. Inspection of bolting material during CRDM maintenance not associated with flange leakage will be performed in accordance with the requirements of Table IWB-2500-1 Item Number B7.80.

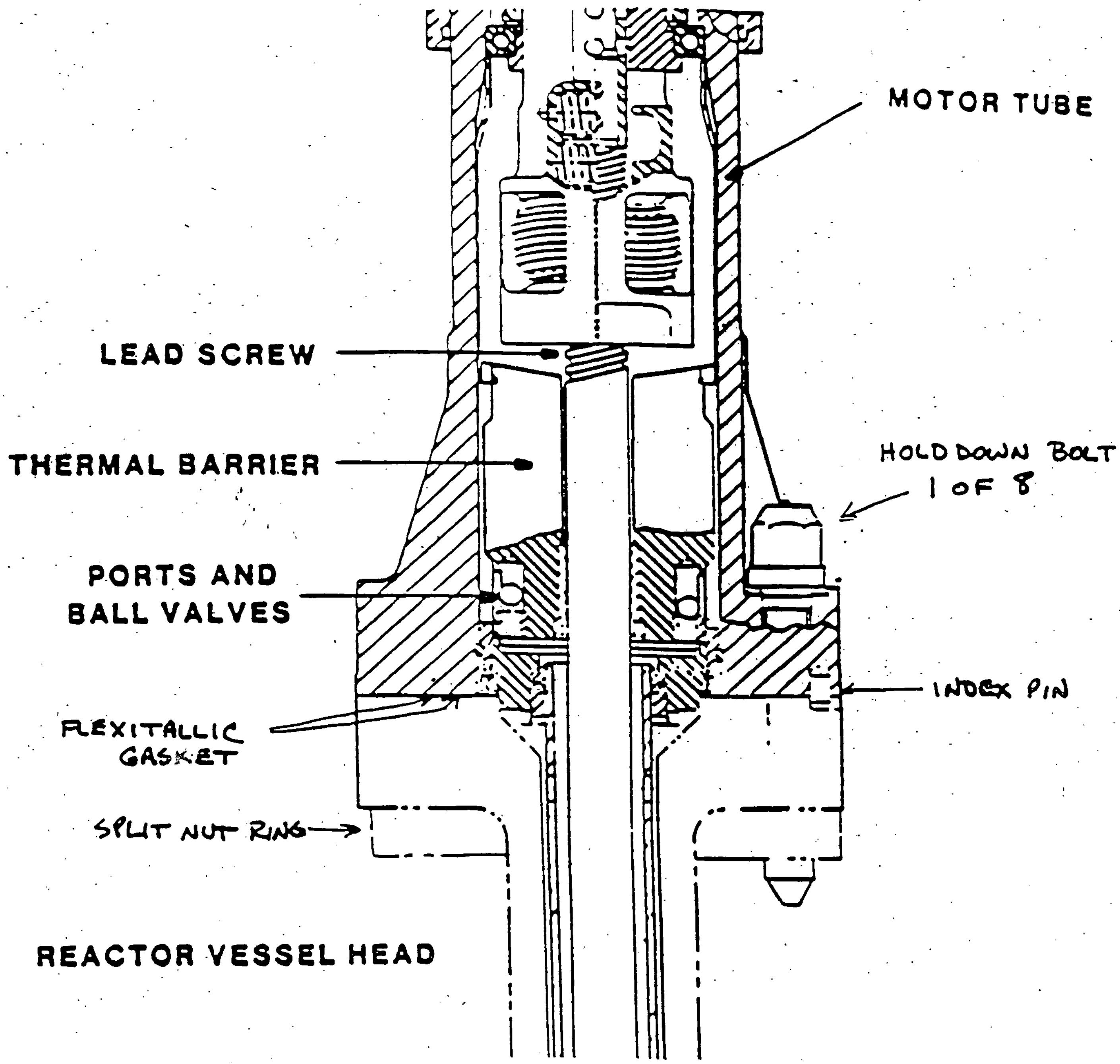
Replacement of bolting material, as a matter of policy, removes the necessity for examination of the material for continued service. Pre-service examination of replacement material ensures the achievement of an acceptable level of safety for the replacement material.

V. Implementation Schedule:

This request was approved for the second ten year inservice inspection interval and will be continued for the third ten year inservice inspection interval for Units 1,2 & 3.

Evaluated By: A. J. Hogge Jr Date 8-18-94

Reviewed By: [illegible] Date 8/23/94



TITLE: CONTROL ROD DRIVE MECHANISM	NOTES: THERMAL BARRIER	OC-PNS-CRD-10, 12-3-86
		Diamond Power 7032551058-D
		DMC / ARB, KPB

DUKE POWER COMPANY

Request for Relief From
Inservice Inspection Requirement

Station: **Oconee**

Unit: **1, 2 & 3**

Requesting Department: Nuclear Generation

Reference Code: ASME Section XI, 1989 Edition, no addenda

I. Component for which exemption is requested:

a. Name and Identification Number:

Reactor Pressure Vessel 36" outlet nozzle-to-vessel welds and outlet nozzle-to-pipe welds (Unit 1 OM-201-5) Attachment ("A"); (Unit 2 OM-1201-4) Attachment ("B"); (Unit 3 OM-2201-52) Attachment ("C").

Oconee 1

Item No.	ID No.	Description
B03.090.001A	1-RPV-WR13	Noz. to Vsl
B03.090.002A	1-RPV-WR13A	Noz. to Vsl
B03.100.001	1-RPV-WR13	Inside Radius
B03.100.002	1-RPV-WR13A	Inside Radius
B09.011.065	1-PHA-1	Noz. to Pipe
B09.011.077	1-PHB-1	Noz. to Pipe

Oconee 2

Item No.	ID No.	Description
B03.090.001A	2-RPV-WR13	Noz. to Vsl
B03.090.002A	2-RPV-WR13A	Noz. to Vsl
B03.100.001	2-RPV-WR13	Inside Radius
B03.100.002	2-RPV-WR13A	Inside Radius
B09.011.019	2-PHA-1	Noz. to Pipe
B09.011.021	2-PHB-1	Noz. to Pipe

Oconee 3

Item No.	ID No.	Description
B03.090.001A	3-RPV-WR13	Noz to Vsl
B03.090.002A	3-RPV-WR13A	Noz to Vsl
B03.100.001	3-RPV-WR13	Inside Radius
B03.100.002	3-RPV-WR13A	Inside Radius
B09.011.001	3-PHA-1	Noz to Pipe
B09.011.003	3-PHB-1	Noz to Pipe

b. Function:

Welded connection between the reactor pressure vessel and respective reactor coolant piping providing a flow path to the steam generator.

c. ASME Section XI Code Class:

Class 1

d. Construction Code and Class (If Applicable):

ASME Section III, 1965 Edition with Summer 1967 Addenda; Class 1

e. Valve Category (If Applicable):

N/A

II. Reference Code Requirement that has been determined to be impractical:

ASME Boiler and Pressure Vessel Code Section XI, 1989 Edition, no addenda, Table IWB-2500-1 (Category B-D), Item Numbers B3.90 and B3.100. NOTE (2): At least 25% but not more than 50% (credited) of the nozzles shall be examined by the end of the first inspection period of each inspection interval.

ASME Boiler and Pressure Vessel Code Section XI, paragraph IWB-2420(a): The sequence of component examinations established during the first inspection interval shall be repeated during each successive inspection interval to the extent practical.

III. Basis for Requesting Relief:

During the first period of the second ten year inspection interval at Oconee Nuclear Station the reactor vessel 36" outlet nozzle-to-vessel welds, including nozzle-to-pipe welds, were examined using Babcock & Wilcox's Automated Reactor Inspection Tool (ARIS). The two nozzle welds examined met the 25% requirement of Table

IWB-2500-1. No recordable indications were detected.

During the third period of the second ten year inspection interval all reactor vessel nozzle-to-vessel and respective nozzle-to-pipe welds were examined using ARIS. Included in this examination was the 36" outlet nozzle-to-vessel and nozzle-to-pipe welds examined during the first period. The re-examination of these 36" outlet nozzles was performed meeting the requirements of the 1989 ASME Section XI Code. Credit will be applied to the third interval, first period requirement for the 36" outlet nozzle-to-vessel welds. Category B-D, Items B3.90 and B3.100. These examinations will not be performed during the first period of the third inspection interval.

Following this inspection sequence will substantially reduce radiation exposure (2 man-rem), critical path time (300 man hours), contaminated shipments, and generation of rad-waste, without effecting the safe operation or reliability of the of the reactor vessel.

IV. Alternate Examination:

Automated re-examination of all the reactor vessel nozzle-to-vessel welds, including respective nozzle-to-pipe welds will be deferred to the last period of the third ten year inspection interval.

V. Implementation Schedule:

Examinations are scheduled to be performed during the third inspection period as follows:

Refueling Outage 21, 2003	(Unit 1)
Refueling Outage 20, 2003	(Unit 2)
Refueling Outage 21, 2004	(Unit 3)

Evaluated By: A. J. Hogge Jr. Date 8-18-94

Reviewed By: J. Barlow Date 8/23/94