NRC FORM 195 (2-76)

U.S. NUCLEAR REGULATORY COMMISSION

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TO:	FROM:	DATE OF DOCUMENT
Mr. A. Schwencer	Duke Power Company Charlotte, North Carolina William O. Parker, Jr.	5/26/77
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ACKNOWLEDGED PLANT NAME: (1-P) Oconee Unit No. 1 RJL DO NOT REMOVE	Consists of requested additional information concerning the Unit No. 1 Inservice Inspection Requirements....... (22-P)

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NRC FORM 195 (2-76)

DUKE POWER COMPANY
POWER BUILDING
422 SOUTH CHURCH STREET, CHARLOTTE, N. C. 28242

May 26, 1977

WILLIAM O. PARKER, JR.
VICE PRESIDENT
STEAM PRODUCTION

TELEPHONE: AREA 704
373-4083

Mr. Edson G. Case, Acting Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. A. Schwencer, Chief
Operating Reactors Branch #1

Reference: Oconee Unit 1
Docket No. 50-269

Dear Sir:

In response to your April 5, 1977 letter which requested additional information concerning the Oconee Unit 1 Inservice Inspection Requirements, the attached information is provided to supplement our October 1, 1976 submittal.

Very truly yours,

William O. Parker, Jr

MST:ge

Attachment

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
CONCERNING THE OCONEE 1 INSERVICE INSPECTION PROGRAM

ATTACHMENT 1 - "Conformance with ASME Section XI, Inservice Examination" Comments

QUESTION

1. Provide additional information using the letter from A. Schwencer, Chief, ORB #1 to W. O. Parker, Jr., Vice President, Steam Production Duke Power Company, dated November 30, 1976, as guidance, to justify the request for relief from the inservice examination requirements specified in the 1974 edition of the Section XI Code through the summer 1975 addenda for Code Class 1, 2 and 3 components.

RESPONSE:

Attachment 1 to our October 1, 1976 submittal described those general exceptions which are taken to the ASME Section XI Code due to anticipated impracticalities which was felt would arise during examination of certain Duke Class B and C components. Any portion for which it is determined that relief from the code requirements is necessary will be submitted to the NRC in accordance with Appendix B of Mr. A. Schwencer's letter to Mr. W. O. Parker dated November 30, 1976. At this time, the only component for which specific relief is requested is the examination of the reactor vessel nozzles. This request for relief is as follows:

1. Component for which relief is requested:

   a. Name and Number: Reactor Pressure Vessel; NRC Docket No. 50-269

   b. Function: Reactor Core Support, Reactor Coolant Pressure Boundary

   c. ASME Section III Code Class: Equivalent Class I per NRC Regulatory Guide 1.26, Revision 2

   d. Valve Category: Not Applicable

2. ASME Section XI requirement that has been determined to be impractical:

   ASME Boiler and Pressure Vessel Code Section XI, 1974 Edition through Summer 1975 Addenda. Paragraph IWB-2411; Subarticle IWB-2500; Table IWB-2500 Category B-D; Table IWB-2600 Item No. B1.4.

3. Basis for Requesting Relief:

   The net effect of the above Code requirements is that four nozzles, of a total of eight, must be examined by the end of 80 months of commercial operation. Due to core support structures design of Oconee 1, only the two reactor coolant outlet nozzles are accessible without removing the core barrel, which in turn requires complete defueling. This requirement is, therefore, considered to be impractical.

4. and 5. Alternate Examination and Implementation Schedule.

The following examination program is proposed in lieu of the Code requirements.

Components to be Examined	Examination Schedule (Elapsed Time Since Commercial Service Date)
1 Reactor Coolant Outlet Nozzle	Approximately 40 months[1]
1 Reactor Coolant Outlet Nozzle	Approximately 80 months[1]
4 Reactor Coolant Inlet Nozzles	Approximately 120 months
2 Core Flooding Nozzles	Approximately 120 months

[1] Different nozzle will be examined each inspection

This program reflects the Reactor Vessel Nozzle examination previously contained in the Oconee Technical Specifications.

QUESTION

2. Clarify whether the required inservice examination will be witnessed or verified by a third party inspector even though South Carolina is not an ASME Code state.

RESPONSE:

The required inservice examinations will not be witnessed by a third party inspector.

ATTACHMENT 2 - "Conformance with ASME Section XI, Subsection IWP, Pump Testing" Comments

QUESTION

1. In order to evaluate the Oconee 1 pump testing program the following information is requested:

   a) A list identifying each pump to be tested by system and application.

   b) The test parameters that will be measured for each pump.

   c) The test intervals, i.e., monthly during operation or only during cold shutdown.

RESPONSE:

The October 1, 1976 submittal provided a listing of safety-related pumps which are considered to be ASME Class 1, 2 or 3 and which are provided on emergency power source. These pumps are to be tested in accordance with IWP to the extent practicable, consistent with the existing design. The attached table provides the listing of pumps to be tested by system and application. The test parameters and test intervals for these pumps are also indicated.

QUESTION

2. When certain parameters are not going to be tested and relief is requested, provide the following information:

   a) Specifically identify the ASME Code requirement that has been determined to be impractical for the pump.

   b) Provide information to support the determination that the requirement in (a) is impractical.

   c) Specify the inservice testing that will be performed in lieu of the ASME Code Section XI requirements that have been determined to be impractical or provide the basis for operation of this pump without this ISI.

   d) Provide the schedule for implementation of the procedure(s) in (c) above.

RESPONSE:

The following relief from the requirements of ASME, Section XI, Subsection IWP is requested:

1. a) Requirement: IWP-3300, IWP-3400 (a) Monthly testing of low pressure injection system pump 1A during normal operation.
   b) Reason: During normal plant operation, the LPI pumps can be tested only in the recirculation mode to the BWST. The "A" pump

can only be tested using a piping line-up which contains a 3 inch section of pipe. This restricts flow to approximately 1150 to 1550 gpm. At this low flow, the installed flow and differential pressure instrumentation does not have sufficient accuracy and the relatively flat pump head curve combines to prevent repeatability of this test.

c) Proposed Testing: During cold shutdowns (or monthly in the event of frequent shutdowns) the "A" pump can be fully tested in Decay Heat Removal model. During normal plant operation, the pump will be operated in recirculation mode monthly for 15 minutes or until vibration readings are taken, whichevery is longer. Since this pump is used primarily during cold shutdown operation, degradation is not expected during periods of power operation.

d) This schedule for testing has been implemented.

2. a) Requirement: IWP-3300 (Table IWP-3100-1) Flow Measurement. For Low Pressure Service Water (LPSW) pump "C".

b) Reason: The LPSW pumps supply two headers, LPA and LPB. The LPB header contains only E.S. components which are equipped with flow transmitters. LPA supplies both ES components and a subheader which supplies Non-ES auxiliary equipment which can be isolated only if both Units 1 and 2 are at cold shutdown. The output of pumps A & B may be measured by isolating the two headers so that the entire output of the pump being tested goes into "B" header while "C" pump supplies "A" header. However, "C" pump cannot be isolated from "A" header while operating.

c) Proposed Testing: All other parameters will be tested on "C" pump. The ability of "C" pump to supply the normal requirements of "A" header (which is approximately the same as the ES flow) will verify the general performance of the pump.

d) This testing procedure has been implemented.

3. a) Requirement: IWP-3300 (Table IWP-3100-1) Suction pressure measurement for Spent Fuel Pool Cooling, concentrated Boric Acid, and Low Pressure Boric Acid pump.

b) It is not considered practical to perform these suction pressure measurements since the necessary instrumentation does not exist.

c) Proposed Testing: Level indications exist for the pool/tanks which supply these pumps. These levels, along with known static head differences from reference levels to pump suctions will provide an approximate indication of the pump suction pressure. Velocity losses should be relatively constant from test to test due to the repeatability of flow rates and valve positions for the test.

d) This test procedure has been implemented.

4. a) Requirement: IWP-3300 (Table IWP-3100-1) Flow for Concentrated Boric Acid and Low Pressure Boric Acid Pumps.

b) Reason: It is not practical to perform these measurements since flow measurement devices do not exist in these lines.

c) Proposed Testing: None possible for this parameter.

5. a) Requirement: IWP-3300 (Table IWP-3100-1 footnote 2), Inlet Pressure Pi, for all pumps which are in operation on a routine basis at the time the test is started.
   b) Reason: Several systems are normally in operation with one or more pumps running. Taking inlet pressure prior to pump startup would require an additional transfer to another pump. This (1) increases the time required for the test, (2) causes additional wear on the pumps due to extra starts, (3) on some systems this will require additional radiation dose during valve lineups prior to swap-over, and (4) presents additional opportunity for human error during transfers.
   c) Proposed Testing: Inlet pressure will be taken prior to startup of any standby pumps. Since in most systems standby and operating pumps are alternated periodically, all pumps will be checked periodically. Also, on systems where the inlet piping is common, the operational pump will affect the inlet pressure of the standby pump so that operating pressure on one pump would be the same as pre-start pressure on the standby pump.
   d) This testing procedure has been implemented.

6. a) Requirement: IWP-3300 (Table IWP-3100-1), Lube Oil Level for HPI, CBAT and LP Boric Acid Pumps.
   b) Reason: No indication exists to verify lube oil level without partial disassembly of the pump.
   c) Proposed Testing: None on this parameter.

7. a) Requirement: IWP-3210, (Table IWP-3200-2) Allowable Ranges of Test Quantities.
   b) Reason: In reviewing Section IWP-4100, a general discrepancy was noticed. This is that IWP-4111 and Table IWP-4110-1 specify that an instrument full scale range may be four times the reference value with nominal errors (in most cases) of $\pm$ 2% of full scale. This permits an error range of $\pm$ 8% of the reference value. By Table IWP-3100-2, flow and pressure readings are allowed to range only +2, -6% (flow) and +2, -7% (pressure). Therefore, a test could fail (exceeding the +3% required to enter the Required Action Range) entirely due to instrument error. Even recalibration per IWP-3230(b) would not help if the instrument was still within its $\pm$ 2% full scale accuracy.
   c) Proposed Testing: Therefore, it is requested that per IWP-3210 the ranges on flow and pressure be extended as shown:

Test Quantity	Acceptable Range	Alert Ranges		Required Action Range	
		Low Values	High Values	Low Values	High Values
ΔP	.92 to 1.08ΔPr	.90 to .92ΔPr	1.08 to 1.10ΔPr	<.90 Pr	>1.10ΔPr
Q	.92 to 1.08 Qr	.90 to .92 Qr	1.08 to 1.10 Qr	<.90 Qr	>1.10 Qr

Also, it is requested that relief from Table IWP-4110-1 be granted for the installed plant instrumentation listed below which have nominal tolerances greater than $\pm$ 2% of full scale:

Instrument Number	IWP Pump	IWP Parameter	Existing Tolerance
Ft-75	HPI Pumps	Flow	± 2.5%
Ft-2A	R.B. Spray	Flow	± 3%
Ft-3A	R.B. Spray	Flow	± 3%
Lt-1	Spent Fuel Cooling	Level for suction pressure	± 2.7%
Pt-176	CBAT Pump	Disch. Press.	± 1.6% with span greater than 4 times the reference value

Attachment 2 Comments Item 1 Pumps	Frequency	Unit Status	Inlet Pressure	Diff. Pressure	Flow	Vibration	Lube Oil Level	Bearing Temperature	Shaft Speed
Item High Pressure Injection Pumps (1A, 1B, 1C) Primary Make-up	Mo 1	HS AP	X	X	X	X	2	X	X
Low Pressure Injection Pumps Decay Heat Removal 1A, 1B, 1C	MO	2 NA	X	X	X	X	X	X	X
Reactor Building Spray Pumps 1A, 1B	MO 1	HS AP	X	X	X	X	X	X	X
Low Pressure Service Water Pumps - 1A, 1B, 1C Services Decay Heat Coolers Reactor Building Vent. Coolers	MO	NA	X	X	2	X	X	X	X
Spent Fuel Pool Cooling Pumps 1A, 1B	MO	NA	2	2	X	X	X	X	X
Emergency Feedwater Pump	MO	NA	X	X	X	X	X	X	X
Concent. Boric Acid Pump 1WD-P22 Boric Acid Addition	MO	NA	2	2	2	X	2	X	X
Low Pressure Boric Acid Pumps 1A, 1B, Boric Acid Addition	MO	NA	2	2	2	X	2	X	X

Notes:

1. HPI and RB Spray pumps cannot be operated at cold shutdown. Therefore, per IWP-3400 (a), they will be tested within 7 days after any cold shutdown which coincides with the due date of the test.

2. See attached list of requested exemptions for exceptions.

ATTACHMENT 3 - "Conformance with ASME Section XI, Subsection IWV, Valve Testing" Comments

QUESTION

1. Provide code class designations for all valves tested.

2. On your table identify each valve in ASME Section XI Cat. A that will be leak tested during refueling outages.

3. Provide the test intervals for all valves to be tested. For check valves, identify those that will be exercised only during cold shutdown.

RESPONSE:

The October 1, 1976 submittal provided a listing of those safety-related ASME Class 1, 2 and 3 valves for which testing is considered to be required. This listing was provided to meet the requirements of Section IWV-1400 which requires the owner to identify the specific valves to be tested. Since no differentiation in testing requirements exist for various ASME class valves, the specific code class designation of each valve is unnecessary.

Valves which are classified as ASME Section XI Category A valves are indicated on the table of valves provided in the October 1, 1976 submittal. All valves which require leak tests will be tested annually, most probably during the refueling outage.

Valves will be tested at the frequency required by ASME Section XI, Subsection IWV unless specifically identified on the valve table. Please note a corrected copy of the valve list is attached.

With regard to check valves, the testing frequency will be at least quarterly unless indicated that it cannot be performed during power operation. In these cases, pursuant to IWV-3520(b) these valves will be exercised during each plant cold shutdown but not more often than once every 9 months.

QUESTION

4. Where relief has been requested from certain requirements of the code, specify the inservice testing that will be performed in lieu of the ASME Code Section XI requirements that are impractical or provide the bases for operation of this valve without this ISI. Also provide the schedule for implementation of this testing.

RESPONSE:

The valve listing attached identifies those valves for which certain requirements of the code are impractical. Valves with a comment code "1" are to be tested at time other than power operation as is permitted by Subsections IWV-3410 and IWV-3520. Those valves with a comment code "2" are impractical to leak test due to the lack of appropriate test connections or isolation valves. Those valves with a comment code "3" are impractical to exercise test. No testing will be performed in lieu of the ASME Code Section XI requirements that are impractical.

The intent of 10CFR50.55a is to require that throughout the service life of a nuclear facility the inservice inspection program shall meet the requirements of Section XI of editions of the ASME Boiler and Pressure Vessel Code and Addenda which become effective to the extent practical. It is our understanding that the code does not require upgrading of the design of the facility, but rather where practical, to improve the inspection or testing criteria or methods. In the case of Subsection IWV for valve testing, certain provisions of the code have been identified which are not practical to meet. These primarily are the results of insufficient test connections or isolation valves to enable leak tests to be performed; inappropriate piping configurations to permit exercise testing of check valves; or unaccessibility of components to verify operation of the valve to be tested. Continued operation without inservice inspection of these components is not considered to be detrimental to the public health and safety over the "as licensed plant" for the following reasons: Many of the valves are subjected to high system pressures during normal operation and unacceptable leakage would be readily detected, e.g., core flood tank valves CF-3, CF-4, CF-19, CF-33, etc. Some valves which cannot be tested perform containment isolation functions, however, the downstream piping is adequately designed for accident conditions, e.g., HP-20 reactor coolant pump seal return; many valves which cannot be specifically tested are but one of two redundant isolation valves. Additionally, all systems are functionally tested during the periodic containment integrated leak rate test to provide assurance of operability. In consideration of the burden which would be imposed to enable testing of these components in accordance with the code, it is not felt that the health and safety of the public would be significantly improved.

Additionally, the following specific relief from the code is requested:

1. a) Requirement: IWV-3410(c) Power operated valves.
   b) Reason: Power operated valves which operate in very short time periods (in the order of one second) are difficult to accurately time. In these instances, the specified limiting valve of the full stroke time will generally be considerably greater than the actual full stroke time. In accuracies in timing contribute to not being able to meet the acceptance criteria of IWV-3410(c)(3).
   c) Proposed testing: If any valve with a previously measured stroke time less than or equal to one second is observed to increase in stroke time to slower than 1.5 seconds, test frequency shall be increased to once each month until corrective action is taken, at which time the original test frequency shall be resumed. In any case, any abnormality or erratic action shall be reported."

QUESTION

5. Provide simplified piping diagrams of systems which must function to safely shutdown the plant or mitigate the consequences of an accident. Active components on the above systems which must change position should be identified. Also, provide a narrative description of the valve line-ups required of the systems identified above each of their safety functions.

RESPONSE:

P & ID draings have been supplied for review. These drawings indicate active components and should provide adequate information for personnel familiar with operation of a B&W steam supply system to perform this review. If assistance is required in understanding the operation of these systems, this could more readily be resolved in verbal discussions than in the requested narrative descriptions.

QUESTION

6. In addition to the above comments, we have found that in some cases, valves important to plant safety were omitted. Comments on these valves will be made by drawing number.

   a) Dwg PI-100-A-1 - Only 2 of the three pressurizer relief valves listed. Valve # 1RC-66 should be included.

   b) Dwg PO-101A-1 - Valves listed on this drawing are containment isolation valves. Check valve #HP-194 should be included if possible to the test program.

   c) Dwg PO-102A-1 - Valves BS-3 and BS-4, Power operated valves on the suction lines to the reactor building spray pumps are omitted. They should be included or justification provided for not including them.

   d) Dwg PO-103A-1,2,3 - Check valves BS-14 and BS-19 are proposed to be tested every 5 years. A source of instrument air exists (according to the drawing) for spray nozzle testing. The licensee should consider using the instrument air to test the check valves on a more frequent schedule.

   e) Dwg PO-122A-1 - Only one of sixteen main steam safety reliefs is listed. Provide justification for not including the others.

   f) Dwg PO-127-B - We cannot locate N2 isolation valves 1N-91 thru 1N-94. These may be mis-numbered on submittal. They should be N-105, 106, and 107. Confirm and correct technical specification as required.

RESPONSE:

   a) Valve 1RC-66 is the power operated relief valve as distinguished from the other two code relief valves. This valve performs no specific safety function and does not have specific leakage requirements other than the Technical Specification primary coolant leakage requirements. It is not considered that this valve should require testing.

   b) Check valve HP-194 has been included in the test program as indicated in the attached revised table.

   c) Power operated valves BS-2 and BS-4 have been included in the test program as indicated in the attached revised table.

d) The current Oconee Technical Specifications require the testing of valves BS-14 and BS-19 at five year intervals. Due to the difficulty in obtaining accessibility in verifying instrument air flow through the spray nozzles, it is considered that this test is not practical at more frequent intervals. Since these valves are not subjected to liquids and are not in a corrosive atmosphere, it is considered that this test interval is satisfactory.

e) All sixteen main steam safety valves will be tested as indicated on the revised valve table.

f) Valves 1N-91 thru 1N-94 were included by error. The correct valves should be 1N-106 and 1N-107. Valve 1N-105 does not require testing as it performs no isolation function.

ATTACHMENT 4 - "Proposed Technical Specification Revision, Inservice Inspection" Comments

QUESTION

1. Because of difference in the date of start of facility commercial operation for Oconee Units 1, 2 and 3, we recommend that separate Technical Specification be established for each unit.

RESPONSE:

Oconee Unit 1 was required to conform to the provisions of 10CFR50.55a beginning November 15, 1977. Oconee Units 2 and 3 will be required to meet these criteria on January 9, 1978 and April 16, 1978, respectively. The October 1, 1976 submittal provided proposed Technical Specifications amendments which were designed to separate the surveillance requirements of Oconee 1 and Oconee 2,3. In the future, after Oconee 2 and 3 meet the criteria of 10CFR50.55a, the Technical Specifications language should be general enough to permit common usage for all three Oconee units.

QUESTION

2. The language in the Technical Specifications 4.04 and 4.2.1 is not acceptable. The sample technical specification language recommended in the letter from R. A. Purple, Chief, Operating Reactors Branch #1, NRC to Duke Power Company, dated April 26, 1976 should be used, i.e., 4.2.1 - Inservice inspection of ASME Code Class 1, 2 and 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10CFR50, Section 50.55a(g), except where specific written relief has been granted by the NRC pursuant to 10CFR50, Section 50.55a(g)(6)(i).

   4.0.4 - Inservice testing of ASME Code Class 1, 2 and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10CFR50, Section 50.55a(g), except where specific written relief has been granted by the NRC pursuant to 10CFR50, Section 50.55a(g)(6)(i).

RESPONSE:

The provisions of 10CFR50.55a(g)(i) require that Oconee 1 meet the inservice inspection requirements of paragraphs (g)(4) and (g)(5) to the extent practical. Paragraph (g)(4) requires that "components classified as ASME Code Class 1, 2 and 3 shall meet the requirements except design and access provisions and preservice examination requirements ......that become effective subsequent to editions......in paragraph (b) of this section to the extent practical within the limitations of design, geometry and materials of construction of the components".

Paragraph (g)(5)(i) requires the licensee to revise the inservice inspection program, as necessary, to meet the provisions of paragraph (g)(4).

Paragraph (g)(5)(ii) requires that License Amendments be submitted at least six months before the start of a period if the revised inservice inspection program conflicts with the Technical Specifications of the facility.

Paragraph (g)(5)(iii) makes provisions for the licensee to notify the Commission if it is determined that conformance with certain code requirement is impractical.

Finally, Paragraph (g)(5)(iv) requires that where an examination or test requirement by the code is determined to be impractical by the licensee and is not included in the revised inservice inspection program, the basis shall be demonstrated to the satisfaction of the Commission not later than 12 months after the expiration of the initial 120 month period of operation.

As can be seen from the above summarization of the regulations, the licensee is required to perform an inservice inspection program in accordance with the ASME Code which becomes in effect to the extent practical within the limitations of design, geometry, and materials of construction of the components. If conflicts exist between the inservice inspection program and the Technical Specifications, a license amendment resolving the conflict shall be submitted six months prior to the start of the inspection interval. Further, if it is determined that conformance with certain provisions of the code are impractical, the licensee shall notify the Commission and shall demonstrate this to the satisfaction of the Commission within 11 years from the start of the inspection interval.

It is our understanding of the regulations that Commission approval of the inservice inspection program is not required prior to the start of an interval unless a conflict exists with the Technical Specifications. Specific written relief must be granted by the NRC within 11 years. Therefore, the proposed Oconee Technical Specifications which cite the regulations and require that the testing and examinations be performed to the extent practicable within the limitations of design, geometry and materials of construction are adequate and properly implement the regulation. Additionally, if the standard wording were adopted and written relief was not received from the Commission prior to the start of the interval, we would be in a situation of noncompliance.

It should also be noted that the regulations do not require submittal of a description of the inservice inspection program every 40 months and the pump and valve testing every 20 months unless specific relief is requested. It is considered that the information requested by Appendix A to Mr. A. Schwencer's letter dated November 30, 1977 is unnecessary after the initial submittal.

QUESTION

3. Technical Specifiation 4.2.6 regarding the pump flywheel inservice inspection program is not acceptable. We require that a surface examination of all exposed surfaces and a complete volumetric examination, during the plant shutdown coinciding with the inservice inspection schedule as required by the Section XI Code be performed at approximately ten-year intervals, in addition to the in-place volumetric examination of the bore and keyway of each reactor coolant pump flywheel at approximately three-year intervals as specified in Technical Specification 4.2.6. Removal of the flywheel is not required.

RESPONSE:

The following wording is proposed for the pump flywheel inservice inspection program:

4.2.6 Reactor coolant pump flywheels shall be examined as follows. An inplace volumetric examination of the bore and keyway shall be performed at approximately three-year intervals coinciding with ASME Section XI inservice inspection examinations. Additionally, a surface examination of all exposed surfaces and a complete ultrasonic volumetric examination of the flywheel shall be conducted at approximately ten-year intervals coinciding with ASME Section XI Inservice inspection examinations.

OCONEE UNIT 1
CONFORMANCE WITH ASME SECTION XI, SUBSECTION IWV

DRAWING NO. Valve No.	Valve Name	Type	Leak Test	Exercise Test	Safety Valve Test	Check Valve Test	Category	Lock Open /Closed	Comments
PO-100A-1									
1RC-67	Pressurizer Relief	R			X		C		
1RC-68	Pressurizer Relief	R			X		C		
PO-101A-1									
1HP-24	A HPI Pump Suct. From BWST	P		X			B		
1HP-25	C HPI Pump Suct. From BWST	P		X			B		
1HP-101	A HPI Suct. Check Vlv.	C				X	C		1
1HP-102	C HPI Suct. Check Vlv.	C				X	C		1
1HP-105	A HPI Disch. Check Vlv.	C				X	C		
1HP-109	B HPI Disch. Check Vlv.	C				X	C		
1HP-113	C HPI Disch. Check Vlv.	C				X	C		
1CA-85	BAMT to LDST	C				X	C		
1CA-73	CBAST to LDST	C				X	C		
1HP-16	Makeup to LDST	P		X			B		
1LP-57	LPI to HPI C Train	C				X	C		1
1LP-55	LPI to HPI B Train	C				X	C		1
PO-101B-1									
1HP-3	A LD Cooler Outlet	P	X	X			A/B		
1HP-4	B LD Cooler Outlet	P	X	X			A/B		
1HP-5	LD Cooler Isolation	P	X	X			A/B		
1HP-20	RC Pump Seal Return	P	X	X			A/B		1,2
1HP-21	RC Pump Seal Return	P	X	X			A/B		1
1HP-26	A Loop Injection	P		X			B		
1HP-27	B Loop Injection	P		X			B		
1HP-188	B Loop Check Valve	C				X	C		1
1HP-153	B Loop Check Valve	C				X	C		1
1HP-152	B Loop Check Valve	C				X	C		1
1HP-194	A Loop Check Valve	C				X	C		1
PO-102A-1									
1CF-3	A CFT Sample/Drain	P	X				A		2
1CF-4	B CFT Sample/Drain	P	X				A		2

OCONEE UNIT 1
CONFORMANCE WITH ASME SECTION XI, SUBSECTION IWV

DRAWING NO. Valve No.	Valve Name	Type	Leak Test	Exercise Test	Safety Valve Test	Check Valve Test	Category	Lock Open /Closed	Comments
1CF-7	CFT to MWHUT	M	X				A		2
1CF-1	A CFT Isol Valve	P					E	X	
1CF-2	B CFT Isol Valve	P					E	X	
1CF-5	A CFT Vent	P					E	X	
1CF-6	B CFT Vent	P					E	X	
1CF-11	A CFT Disch Check Vlv.	C				X	C		1
1CF-12	A CFT Disch Check Vlv.	C				X	C		1
1CF-13	B CFT Disch Check Vlv.	C				X	C		1
1CF-14	B CFT Disch Check Vlv.	C				X	C		1
1CF-19	CFT to Sample Sink	M	X				A		2
1CF-33	CFT Vent to Vent Hdr.	M	X				A		2
1CF-35	CFT Vent to WG Filter	M	X				A		
1CF-36	CFT Vent to Vent Hdr.	M	X				A		
1LP-1	DH Isolation Valve	P		X			B		1
1LP-2	DH RB Isolation Valve	P		X			B		1
1LP-3	DH RB Isolation Valve	P		X			B		
1LP-6	LPI Suction XConn	P		X			B		
1LP-7	LPI Suction XConn	P		X			B		
1LP-9	LPI Disch XConn	P		X			B		
1LP-10	LPI Disch XConn	P		X			B		
1LP-12	LPI A Cooler Outlet	P		X			B		
1LP-14	LPI B Cooler Outlet	P		X			B		
1LP-17	LPI A RB Isol. Vlv.	P		X			B		
1LP-18	LPI B RB Isol. Vlv.	P		X			B		
1LP-19	RB Emerg. Sump	P		X			B		
1LP-20	RB Emerg. Sump	P		X			B		
1LP-21	BWST to LPI Suct.	P		X			B		
1LP-22	BWST to LPI Suct.	P		X			B		
1LP-29	BWST to A LPI Hdr.	C				X	C		
1LP-30	BWST to B LPI Hdr.	C				X	C		
1LP-31	A LPI Pump Disch	C				X	C		
1LP-33	B LPI Pump Disch	C				X	C		
1LP-51	Caustic Addn.	M		X			B		
1LP-28	BWST Isolation	M					E	X	

OCONEE UNIT 1
CONFORMANCE WITH ASME SECTION XI, SUBSECTION IWV

DRAWING NO. Valve No.	Valve Name	Type	Leak Test	Exercise Test	Safety Valve Test	Check Valve Test	Category	Lock Open /Closed	Comments
1LP-47	A LPI Hdr. Check Vlv.	C				X	C		1
1LP-48	B LPI Hdr. Check Vlv.	C				X	C		1
1LP-103	Boron Dilution Vlv.	P		X			B		1
1LP-104	Boron Diltuion Vlv.	P		X			B		1
1LP-105	Boron Dilution Vlv.	P		X			B		1
1LP-15	LPI A Hdr. to HPI	P		X			B		
1LP-16	LPI A Hdr. to HPI	P		X			B		
1BS-5	A RBS Check	C				X	C		
1BS-6	B RBS Check	C				X	C		
1BS-7	A LPI Hdr. to RBS	C				X	C		
1BS-9	B LPI Hdr. to RBS	C				X	C		
1BS-3	A RBS Suct.	P		X			B		
1BS-4	B RBS Suct.	P		X			B		
PO-103A-1									
1BS-1	A RBS RB Isol. Valve	P		X			B		
1BS-2	B RBS RB Isol. Valve	P		X			B		
1BS-11	A RBS Disch. Check	C				X	C		
1BS-14	A RBS Disch. Check	C				X	C		1 Will be tested every 5 yrs.
1BS-16	B RBS Disch. Check	C				X	C		
1BS-19	B RBS Disch. Check	C				X	C		1 Will be tested every 5 y
PO-104A-1									
1SF-60	Fuel Transfer Canal Fill	M	X				A		2
1SF-61	Fuel Transfer Canal Fill	M	X				A		2
PO-106-A-1									
1CS-64	CBAST Outlet	P		X			B		
PO-106E-1									
1FW-64	Filtered Water to RB	M	X				A		2
1FW-65	Filtered Water to RB	M	X				A		2
1DW-155	DW to RCP Seal Vent	C	X			X	A/C		1,2,3
1DW-156	DW to RCP Seal Vent	C	X			X	A/C		1,2,3
1DW-59	DW to RB	M	X				A		2

OCONEE UNIT 1
CONFORMANCE WITH ASME SECTION XI, SUBSECTION IWV

DRAWING NO. Valve No.	Valve Name	Type	Leak Test	Exercise Test	Safety Valve Test	Check Valve Test	Category	Lock Open /Closed	Comments
1DW-60	DW to RB	M	X				A		2
PO-107A-1									
1CS-5	QT RB Isol.	P	X	X			A/B		2
1CS-6	QT RB Isol.	P	X	X			A/B		
1CS-12	QT Recirc. Check	C	X			X	A/C		1,3
1CS-11	QT Recirc. Check	C	X			X	A/C		3
1GWD-12	QT Vent	P	X	X			A/B		2
1GWD-13	QT Vent	P	X	X			A/B		
PO-107B-1									
1LWD-1	Normal Sump Suct.	P	X	X			A/B		
1LWD-2	Normal Sump Suct.	P	X	X			A/B		2
PO-107D-1									
1LWD-99	RB Sump to LAWT	M	X				A		2
PO-110A-1									
1CA-17	BAMT to Makeup Filters	C				X	C		
1CA-18	BAMT to Makeup Filters	M		X			B		
1CA-39	Caustic to LP Suction	M		X			B		
1RC-5	Press. Steam Sample	P	X	X			A/B		
1RC-6	Press. Water Sample	P	X	X			A/B		
1RC-7	Press. Sample	P	X	X			A/B		
1FDW-105	OTSG A Sample	P	X	X			A/B		
1FDW-106	OTSG A Sample	P	X	X			A/B		
1FDW-107	OTSG B Sample	P	X	X			A/B		
1FDW-108	OTSG B Sample	P	X	X			A/B		
PO-116A									
1PR-1	RB Purge Outlet	P	X	X			A/B		
1PR-2	RB Purge Outlet	P	X	X			A/B		
1PR-7	RB Radiation Monitor	P	X	X			A/B		
1PR-8	RB Radiation Monitor	P	X	X			A/B		

OCONEE UNIT 1
CONFORMANCE WITH ASME SECTION XI, SUBSECTION IWV

DRAWING NO. Valve No.	Valve Name	Type	Leak Test	Exercise Test	Safety Valve Test	Check Valve Test	Category	Lock Open /Closed	Comments
1PR-9	RB Radiation Monitor	P	X	X			A/B		
1PR-10	RB Radiation Monitor	P	X	X			A/B		
1PR-6	RB Purge Inlet	P	X	X			A/B		
1PR-5	RB Purge Inlet	P	X	X			A/B		
PO-121A-1									
1FDW-93	EFDW OTSG A	C				X	C		1
1FDW-95	EFDW OTSG B	C				X	C		1
PO-121B-1									
1FDW-101	EFDW to OTSG A	C				X	C		1
1FDW-99	EFDW to OTSG B	C				X	C		1
1FDW-33	EFDW to OTSG A	P		X			B		1
1FDW-36	EFDW to OTSG A	P		X			B		1
1FDW-38	EFDW to OTSG A	P		X			B		1
1FDW-42	EFDW to OTSG B	P		X			B		1
1FDW-45	EFDW to OTSG B	P		X			B		1
1FDW-47	EFDW to OTSG B	P		X			B		1
1FDW-104	OTSG B Drain	P	X	X			A/B		2
1G-23	OTSG B Drain	M	X				A		2
FDW-103	OTSG A Drain	P	X	X			A/B		2
2G-23	OTSG A Drain	M	X				A		2
PO-122A-1									
1MS-1 thru 1MS-16	Main Steam Relief	R			X		C		
PO-124B									
1LPSW-6	LPSW to RCP Oil Coolers	P	X	X			A/B		1
1LPSW-15	LPSW from RCP Oil Coolers	P	X	X			A/B		1
1LPSW-18	LPSW to RBCU A	P		X			B		
1LPSW-21	LPSW to RBCU B	P		X			B		
1LPSW-24	LPSW to RBCU C	P		X			B		
1LPSW-4	LPSW DH Cooler Outlet	P		X			B		
1LPSW-5	LPSW DH Cooler Outlet	P		X			B		
1LPSW-75	LPSW DH Cooler Outlet	C				X	C		
1LPSW-76	LPSW DH Cooler Outlet	C				X	C		
1LPSW-251	LPSW DH Cooler Outlet	P		X			B		
1LPSW-252	LPSW DH Cooler Outlet	P		X			B		
1LPSW-108	RBCU Outlet	M					E	X	

DRAWING NO. Valve No.	Valve Name	Type	Leak Test	Exercise Test	Safety Valve Test	Check Valve Test	Category	Lock Open /Closed	Comments
PO-127B									
1N-106	N_2 Isolation	M	X				A		2
1N-107		M	X				A		2
1N-116		M	X				A		2
1N119		M	X				A		2
1CA-27		M	X				A		2
1CA-29		M	X				A		2
1N-130		M	X				A		2
1N-129		M	X				A		2
PO-137									
1BA-5	BA Isolation Valve	M	X				A		2
1BA-33	BA Isolation Valve	M	X				A		2
PO-144A									
1CC-20	CC to RCP	C	X			X	A/C		1,3
1CC-24	CC to RCP	C	X			X	A/C		1,3
1CC-76	CC to CRD Service Structure	C	X			X	A/C		1,3
1CC-77	CC to CRD Service Structure	C	X			X	A/C		1,3
1CC-7	CC from RCP	P	X	X			A/B		1,2
1CC-8	CC from RCP	P	X	X			A/B		1
0-472									
1IA-90	Inst. Air to RB	M	X				A		2
1IA-91	Inst. Air to RB	M	X				A		2
0-472									
1LRT-24	Leak Rate Test	M	X				A		2
1LRT-25	Leak Rate Test	M	X				A		2
1LRT-38	Leak Rate Test	M	X				A		2
1LRT-39	Leak Rate Test	M	X				A		2
1LRT-17	Leak Rate Test	M	X				A		2

Valve Type	R - Relief valve P - Power-operated valve, electric or pneumatic C - Check valve M - Manual valve
Leak Test	X - Required
Safety Valve Test	X - Required
Check Valve Test	X - Required
Locked Open or Closed	X - Required to be locked open or closed during power operation
Comments	1 - Valve cannot be exercised during power operation 2 - Provisions for leak testing valve do not exist due to piping configuration 3 - Provisions for exercising valve do not exist

RECEIVED DOCUMENT
PROCESSING UNIT

1977 MAY 31 AM 11 44

Central File
50-269
270
287

Duke Power Company

Power Building

422 South Church Street, Charlotte, N. C. 28242

William O. Parker, Jr.
Vice President
Steam Production

Telephone: Area 704
373-4083

May 24, 1977

Mr. Norman C. Moseley, Director
U. S. Nuclear Regulatory Commission
Suite 818
230 Peachtree Street, Northwest
Atlanta, Georgia 30303

Re: Oconee Nuclear Station
Docket Nos. 50-269, -270, -287

Dear Mr. Moseley:

Pursuant to the requirements of Oconee Nuclear Station Technical Specification 6.6.2.2.d, this report is submitted describing a condition in which a measured level of radioactivity exceeded the control level by greater than four times but less than ten times.

On April 25, 1977, analytical results of composited raw water supply grab samples collected in mid-January, February, and March, 1977 were reviewed. A summary of the pertinent results of the radioactivity concentrations in these samples is given below:

Sample Location	Tritium Concentration
004.1 Seneca (Control)	(2.8 ± 0.7) E-7 μCi/ml
006.1 Clemson	(1.8 ± 0.1) E-6 μCi/ml

Tritium concentrations in downstream water samples are dependent upon the tritium concentrations of liquid effluent released from the station. For the period January 1 through March 15, 1977 a total of 601 curies of tritium were released from the station in liquid effluents. The average release rate for the period was 8.2 Ci/day.

Dilution and dispersion of tritium in liquid effluents between Oconee Nuclear Station and the Clemson water intake has been calculated using the equation for instantaneous release taken from the U. S. Geological Survey Paper No. 443-B, "Dispersion of Dissolved or Suspended Materials in Flowing Streams," by Robert E. Glover (1964), p. 5. This equation accounts for longitudinal dispersion only. Conservatism was used in selecting parameters for substitution in the instantaneous release equation to determine the concentration of effluents at Clemson water

MISC
24

intake. These assumptions were (1) the elevation of Lake Hartwell is 654.00 feet, (2) the flow of the Keowee River is 1100 cfs, the yearly average, and (3) an instantaneous release of 8.2 Ci of tritium is made each day several days prior to sampling. The peak concentration which would result at the 006.1 sample point is 4.0E-6 μCi/ml.

This calculated tritium concentration is about a factor of 2.3 greater than the observed value of 1.8E-6 μCi/ml. Therefore, the observed concentration is within the limits of conservative calculated values.

The Final Environmental Statement for Oconee states that "the largest estimates of dose to the individuals from liquid effluents are at Clemson and Pendleton where drinking water is withdrawn from the Keowee River. The radionuclide making the most important contribution to dose at these locations is tritium (more than 50%)." The dose estimate for any individual consuming Clemson water containing 1.8×10^{-6} μCi/ml of tritium is 0.18 mrem/year if these tritium concentrations were maintained over the year. This estimate of dose is less than 0.15% of the dose from natural background and less than 0.04% of the limits of 10CFR20. Therefore, it is concluded that the observed anomalous tritium concentration does not adversely affect public health and safety.

Very truly yours,

William O. Parker

William O. Parker, Jr.

LJB:vr

May 20, 1977

Docket Nos. 50-269
50-270
and 50-287

Duke Power Company
ATTN: Mr. William O. Parker, Jr.
Vice President - Steam Production
Post Office Box 2178
422 South Church Street
Charlotte, North Carolina 28242

Gentlemen:

In our meeting on March 23, 1977, at the Jocassee Dam site near the Oconee Nuclear Station, we stated what we consider to be appropriate actions for Duke Power Company to take to allow us to evaluate the potential seismic hazard at the Jocassee Dam.

We outlined a threefold program: (1) a short term report, (2) a monitoring program, and (3) a longer-term report that will describe the findings of the monitoring program.

It is requested that the short term report be submitted within 60 days of the date of this letter. The contents of this report were discussed with you at the meeting cited above. The enclosure to this letter provides further details of what the short term report should include.

The enclosure also addresses what we consider to be an adequate monitoring and reporting program. This program includes (1) the installation of three permanent seismic monitoring stations and two to four microearthquake recorders to augment the permanent stations, all which we request that you install by August 1, 1977, (2) reports of the results of the monitoring program submitted to the NRC quarterly, and (3) informing NRC by telephone of any unusual seismic activity as soon as possible. The enclosure provides the details of this program.

The longer-term report should be submitted by January 1, 1979, and summarize and discuss the results of the monitoring program through November 1978.

OFFICE➤						
SURNAME➤						
DATE➤						

It is requested that you submit a request for a license amendment to your facility which would incorporate the requirements discussed herein for the monitoring program and the longer-term report. Please respond within 45 days after receipt of this letter.

Sincerely,

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosure:

cc w/encl:
See next page

DISTRIBUTION
Dockets
NRC(3)
LOCAL PDR
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ACRS (16)
TBAbernathy
JRBuchanan
DEisenhut
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OFFICE ➤	DOR:ORB#1	DOR:ORB#1 KRG			
SURNAME ➤	DNeighbors:lb	ASchwencer			
DATE ➤	5/20/77	5/20/77			

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

May 20, 1977

Docket Nos. 50-269
50-270
and 50-287

Duke Power Company
ATTN: Mr. William O. Parker, Jr.
Vice President - Steam Production
Post Office Box 2178
422 South Church Street
Charlotte, North Carolina 28242

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It is requested that you submit a request for a license amendment to your facility which would incorporate the requirements discussed herein for the monitoring program and the longer-term report. Please respond within 45 days after receipt of this letter.

Sincerely,

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosure:
Monitoring & Reporting Information

cc w/encl:
See next page

cc: Mr. William L. Porter
Duke Power Company
P. O. Box 2178
422 South Church Street
Charlotte, North Carolina 28242

J. Michael McGarry, III, Esquire
DeBevoise & Liberman
700 Shoreham Building
806-15th Street, NW.,
Washington, D.C. 20005

Oconee Public Library
201 South Spring Street
Walhalla, South Carolina 29691

SHORT TERM REPORT

This report should address formally the questions raised during the recent meeting at Jocassee Dam.

(a) Seismicity: All seismic observations gathered to date should be provided in an organized manner, including numbers of events recorded, hypocentral data, focal mechanisms where determined, epicenter maps and depth cross-sections, locations and changes in operating stations and descriptions of network capacity. Also included should be a log of water-level fluctuation.

The report on seismic studies prepared by Law Engineering provides adequate seismic information for the time interval mid-October 1975 until late-June 1976. This report should be resubmitted as part of the total report. For the interval from late June 1976 until the present, no seismic data have been presented to us other than informal oral descriptions.

Consequently, the seismic information for this latter interval should be presented in a formal written report treated in all the detail described in the first paragraph of this section.

(b) Geologic Reconnaissance of the Site Area

The reports submitted to date do not appear to be current and should be modified to depict clearly the current understanding of the location of faulting in the vicinity of Lake Jocassee and the Lake Jocassee Dam. The report of Dr. Conn (Engineering Geology of the Keowee-Toxaway Project, of December 1966, and June 1974) discusses faulting in the vicinity of the dam; a clarification of his findings should be provided. The relevant geologic maps of the site and region and an assessment of the age of last movement of faults in the vicinity of the Lake Jocassee Dam should be provided. Typical construction photographs of the dam rock foundation and abutments should also be provided.

(c) In order to evaluate the seismic adequacy of the dam the following information should be provided:

1. The embankment design and specification should be described along with the foundation treatments used;

2. Seepage rates and changes in seepage rates should be described and plotted;

3. Groundwater profiles (phreatic surface) through the abutments and foundation of the dam should be plotted;

4. The ability of the foundation of the dam to resist the effects of potential fault movements should be assessed and reported. Past measurements of the settling, displacement and cracking of the dam should be interpreted to estimate the existing state of strain, particularly in the core of the dam. The additional strain which can safely be tolerated should be estimated and related to the magnitude of potential fault movement;

5. The tolerance of the abutment material to strain and cracking resulting from fault movement should be estimated based on the properties of the saprolites and the magnitude of potential fault movement. If abutment cracking cannot be ruled out then the piping and erosional resistance of the weathered rock should be assessed.

6. A detailed description of the Federal Power Commission monitoring program for seismic safety should be provided. The dam operating plans in the event of significant seismic excitation should be provided together with plans for immediate inspections and readings of critical instruments. In addition, a plan for the prompt and formal involvement of Duke Power Company geotechnical consultants should be developed to assure that evidence detrimental to the safety of Jocassee dam is not overlooked.

MONITORING PROGRAM

At present it cannot be stated that the levels of activity of late 1975-1976 will not resume. Consequently, it is essential to maintain a monitoring network which will provide accurate and timely information concerning size, frequency and hypocentral data for possible seismic activity.

(a) Seismic Stations: Until November 1978, three permanent stations should be operated by Duke Power and recorded at the damsite at that time a decision will be made, based on the level of activity up to November 1978, as to whether to continue the monitoring program. Two to four microearthquake recorders should be used to augment these stations until December 1977. At that time a decision will be made, based on the level of activity during 1977, as to whether to continue operation of the microearthquake recorders. The two stations in addition to SMT should be installed as soon as possible. Suggested locations for these stations are shown in the attached figure. If possible, arrangements should be made with the USGS and USC to incorporate one or all of the Jocassee stations in the South Carolina network. This would allow recording of these stations on the develocorder at USC. If arrangements are made to include the stations in the network, it may be possible to use USGS radio frequencies for radio telemetry.

To improve the timing resolution for the permanent stations recording speeds of 120 mm/min should be used on the helicorders.

(b) Reporting Procedures: Quarterly reports should be provided to the NRC within one month of the end of each reporting period. These reports should include the following:

1. Text: Short report of the general level of seismicity and and any changes in seismicity.

2. Tables:

   a. Catalog of all earthquakes recorded;

   b. List of all hypocenters located (HYPO71 Format);

   c. Operational report:

      i) Location of stations;

      ii) Times of operation of each station, number of days recording for each station, total number of station-days reporting;

      iii) Report of reasons for any station failures;

3. Figures:

   a. Station locations;

   b. Epicenter locations (with magnitude shown by symbol size);

      i) For reporting period;

      ii) Cumulative, from October 1975;

   c. Graphs of daily water level (and daily range), change in water level/day, number of earthquakes/day, energy release/day, all plotted on the same time scale, for the reporting period;

d. Graphs of the parameters in item 3, above, for ten day intervals from October 1975;

e. Cross-sections of earthquake depths (with error bars) along profiles oriented N-S, E-W, NE-SW, NW-SE and any other profiles suggested by the data.

4. Other Information: If sufficient data are available, "b-values" and focal mechanisms should also be determined. Direction of motion at each station should be included in the report for all earthquakes used in focal mechanism determinations. Interpretation of the significance of these parameters is not required from Duke Power Company station data (HYPO71 format).

   A copy of at least one "typical" seismogram should be included with each report to show data quality and type of activity.

   If felt earthquakes occur, intensity surveys should be carried out and summaries of intensity reports and contoured intensity maps should be included in the report.

5. Abnormal Activity: The NRC should be informed by telephone of any unusual activity as soon as possible. Any of the following should be considered unusual activity:

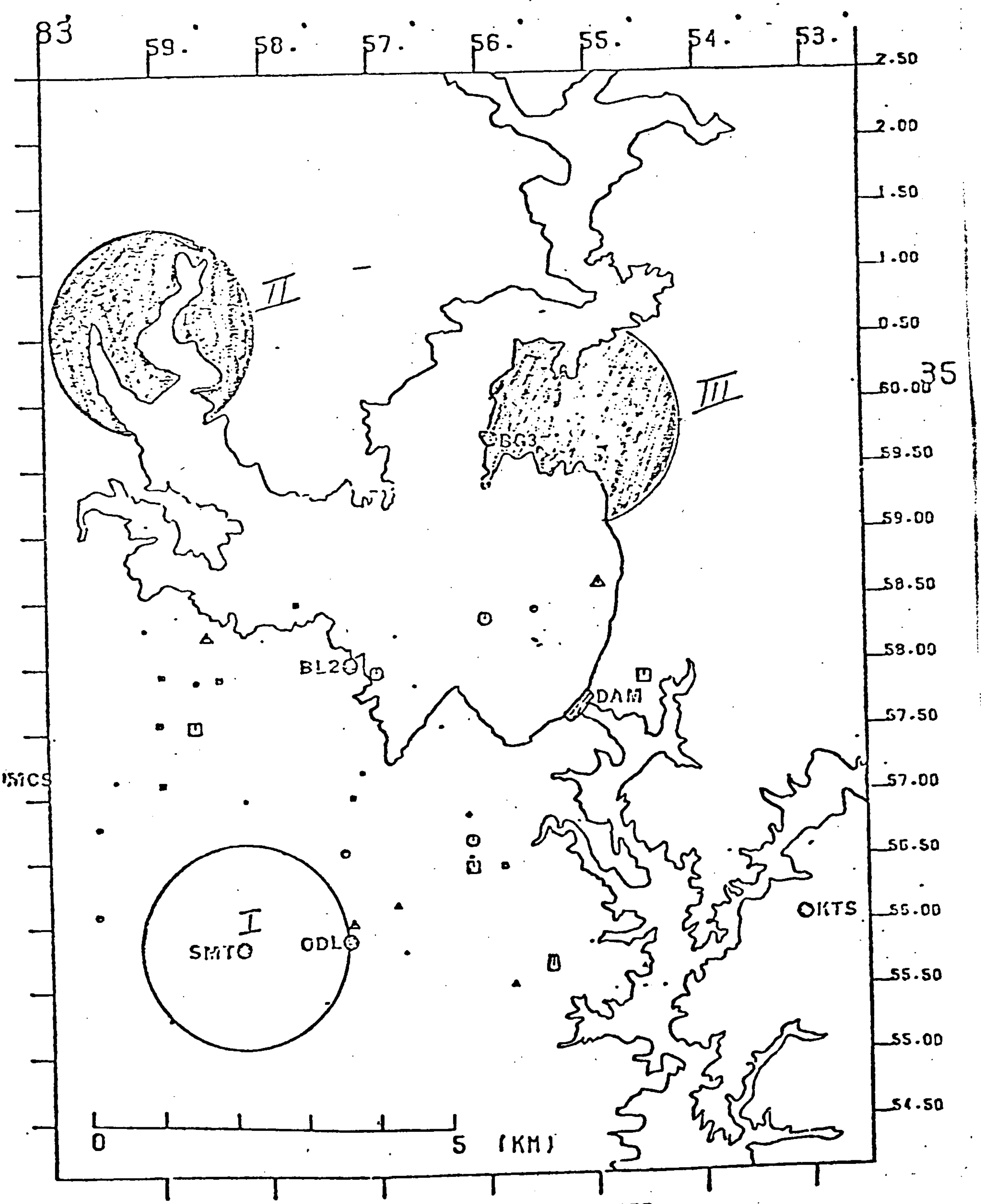
   a. Any earthquake larger than magnitude 2;

   b. More than 100 events per week;

   c. Any plans to make ususual changes in water level in the reservoir.

JOCASSEE EARTHQUAKES

FROM OCT. 28,76 TO FEB. 2,77



RECOMMENDED STATION LOCATIONS AT LAKE JOCASSEE