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

January 11, 1993

Docket No. 50-220

Mr. B. Ralph Sylvia
Executive Vice President, Nuclear
Niagara Mohawk Power Corporation
301 Plainfield Road
Syracuse, New York 13212

Dear Mr. Sylvia:

SUBJECT: RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING 10 CFR 2.206 PETITION SUBMITTED BY BEN L. RIDINGS (TAC NO. M84890)

By letter dated November 19, 1992, we provided Niagara Mohawk Power Corporation (NMPC) with a copy of a Petition submitted to the NRC pursuant to 10 CFR 2.206 by Mr. Ben L. Ridings. Our letter requested NMPC to review the Petition and to provide NMPC's views regarding the Petition. Your letter dated December 21, 1992, provided NMPC's response to our request. Your letter also noted that we had requested Mr. Ridings to submit further information and requested an opportunity to respond to any further submittals.

By letter dated December 4, 1992, to Mr. Ridings, Thomas E. Murley, Director, Office of Nuclear Reactor Regulation, acknowledged receipt of the Petition and requested Mr. Ridings to provide certain information that was not fully legible or not provided in his Petition.

Enclosed is a copy of Mr. Ridings' "Information Requested By Office of Nuclear Reactor Regulation," dated October 27, 1992, which Mr. Ridings has submitted in response to Dr. Murley's December 4, 1992, letter. Mr. Ridings' response is obviously misdated since within his submittal, Mr. Ridings refers to Dr. Murley's December 4, 1992, letter. Mr. Ridings' response was received by the NRC's Office of the Executive Director for Operations on January 5, 1993. In accordance with your request for an opportunity to respond to any further submittals, NMPC is requested to review the enclosed submittal and provide the NRC with NMPC's comments regarding the issues raised in Mr. Ridings' response within 30 days of the date of this letter.

Thank you for your cooperation in this matter. Please contact me at (301) 504-1409 if you have any questions regarding this matter.

This requirement affects one respondent and, therefore, is not subject to Office of Management and Budget review under P.L. 96-511.

Sincerely,

Donald S. Brinkman

Donald S. Brinkman, Senior Project Manger
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosure:
Information Requested
By Office of Nuclear
Reactor Regulation

cc w/enclosure:
See next page

Mr. B. Ralph Sylvia
Niagara Mohawk Power Corporation

Nine Mile Point Nuclear Station
Unit No. 1

cc:

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Power Division, System Operations
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Albany, New York 12223

Ben L. Ridings
P. O. Box 1101
Kingston, Tennessee 37763

MEMO FOR YOUR FILES

Oct 27, 1992

TO: U.S. Nuclear Regulatory Commission
Executive Director for Operations
Public Document Room
1717 H Street
Washington, DC 20555

FROM: Ben L. Ridings
P.O. Box 1101
Kingston, TN 37763

Ref: Petition pursuant 10CFR2.206

Dear Sirs:

Enclosed for filing INFORMATION REQUESTED BY OFFICE OF NUCLEAR REACTOR REGULATION.

Respectfully submitted,

Ben L. Ridings

Ben L. Ridings

Rec'd Off. EDO
Date 1-15-93
Time 9A

UNITED STATED OF AMERICA
BEFORE THE NUCLEAR REGULATORY COMMISSION

INFORMATION REQUESTED BY OFFICE OF NUCLEAR REACTOR REGULATION

Comes Petitioner, in response of the request of the Office of Reactor Regulation:

During the course of review of this petition, keep in mind the FSAR and Test Program being challenged had been reviewed and approved by the NRC. In fact, several revision of these documents had been reviewed and approved by the NRC, Quality Assurance Groups, Utility Management and numerous contracted reviews. This Petitions claims as fact that all groups with responsible administrative duties reviewed the Nine Mile Point Test Program and accepted it as safe and proper to meet the requirements of 10CFR50. The letter to Petitioner, dated December 4, 1992, states in March 1988 the NRC (previous knowledge) identified the administrative deficiencies as defined by Petition. The Petitioner states that in 1990, while the plant was still fully operational, the following existing contradictions in the 50-220 license itself:

Limiting Review to Containment Isolation Valves

1) 6 valves FSAR requires these valve to go open on RPS signal yet it is not mention in TS Table 3.3.4 or FSAR Table VI-3a.
2) 4 valves FSAR VI-3b show valves receive no RPS signal while TS 3.2.7 show valves receive signal to open.
3) 2 valves FSAR gives 10 sec stroke time, TS has 18 sec stroke time.
4) 4 valves FSAR shows RPS logic to close while TS does not
5) 2 valves P&ID's show RPS logic yet not listed in TS or FSAR
6) 8 valves Primary Containment Isolation valves not listed in TS 3.3.4
7) 8 valves FSAR shows RPS logic to close while TS does not
8) 14 valves P&ID show RPS logic while TS and FSAR do not
9) 8 valves FSAR shows RPS logic while TS does not
10) 3 valves P&ID shows HPCI logic while TS and FSAR do not
11) 4 valves FSAR shows RPS logic while TS does not
12) 4 valves FSAR shows these valves as both criterion 56 & 57 valves
13) 22 valves FSAR & TS show these valves as Cat A yet are not LLRT tested

. After twenty years of operation and literally thousands of reviews by "qualified personnel" as stated in the Petition, 45% of the containment isolation valves currently had discrepancies in the license itself. How could this plant be properly built if the license itself contradicted itself. Did these systems have proper RPS logic installed when the guiding design document had these types of discrepancies. How were work packages, procedures or administrative limits reviewed as satisfactorily fulfilled when the review documents contradicted themselves? This is the type of review previously approved adequate by all responsible parties above mentioned. These types of unresolved problems existed after twenty years of "NRC Review and approval".

In Jan 1990, Niagra Mohawk was served the attached memo identifying Category A valves in the FSAR and TS. The leakage rate of each of these valves must be added to the leakage total for containment building. Just as stated in the Petition, when these leakage rates are added to the running total for containment, the facility will no longer meet the leakage total for containment integrity. Secondly, these valves were not simply overlooked. These isolation valves have been purposely placed into closed loops to avoid the addition of these required leakage rates to the running containment leakage total. Some of these so called closed loops are located outside of containment. Miles of RHR Cross-tie piping which are part of the closed loops located outside of containment do not have the same barrier protection for these ECCS systems. As currently approved by the above mentioned review groups, this plant currently has more primary coolant piping outside of containment than it has inside the containment building. The significant increase of exposed piping significantly increases the possibility of a piping shear accident. As stated in the Petition, in order to limit these valves from the leakage total, all responsible parties have now extended the containment miles outside of the containment

barrier. The current administrative controls have allowed this plant to operate outside the minimum required containment leakage total for twenty years. When containment integrity cannot be verified, then public safety is endanger and an immediate action on the part of the regulatory body is demanded. As stated in the Petition, a proper review of TS 4.0.5 and the LLRT program will give an indication of type of review that exists at Nine Mile Point today. Note 17 of the Petition refers to valves identitified in FSAR VI-3b as lines entering free space of containment while TS 3.3.4 identify these valves as Criterion 56.

Safety is not a convenience. When minimum safety requirements are ignored, exempted, or justified as not required in this instance by the executors of nuclear regulation then the fabric of administrative control is torn. As described in the Petition, mandatory safety systems have been justified as not required by the review groups. These same review groups have been wrong and wrong and wrong and wrong. Here, the review groups have stated that Nine Mile Point is not responsible to meet the requirements of 10CFR50 Appendix A-General Design Criteria, justifying this opinion based on an exemption to plants licensed prior to 1974 listed in 10CFR50.46(a)(2). However reading further under this same section [50.46(d)],"the requirements of this section are in addition to any other requirements applicable to ECCS set forth in this part". Further, 10CFR50 Appendix A, clearly states "the General Design Criteria establish minimum requirements for water cooled nuclear power plants for which construction permits have been issued by the commission. Establish the necessary design criteria that provide reasonable assurance that the facility can be operated with no undue risk to the public". The letter to Petitioner, dated Dec 4, 1992, states that at Nine Mile Point the ADS valves could depressurize from reactor pressure to 350 psi so that Core Spray can be used to maintain coolant temperature. This is normal light water plant design.

The problem that exist is makeup water (Criterion 33) which also requires onsite emergency power and a safeguard system to provide proper reactor water level. FSAR VII clearly states "in order to prevent cladding temperatures from exceeding their maximum limit for the entire spectrum of breaks, the 3800 gpm (from one train of HPCI) would have to be available immediately'. Without HPCI availability the possibility of fuel cladding exists.

One must meet the requirements of 10CFR50 in order to operate a commercial nuclear plant in the United States with limited liability. In order to operate Nine Mile Point One under the pretense of limited liability the requirement of 10CFR50 must be meet. This includes the general criteria for design-Appendix A. An ECCS HPCI Safety System is required for insurance with the pretense of limited liability. The letter to Petitioner date Dec 4, 1992 clearly states no such system exists at Nine Mile Point. "The commission stressed that the GDC were not new requirements and were promulgated to more clearly articulate the licensing requirements and practice in effect at that time". Safety is not a convenience but a duty to public safety.

It is Congress's duty to protect public safety and its current administrative controls have failed.

Respectfully submitted,

Ben L. Ridings

Ben L. Ridings
P.O. Box 1101
Kingston, TN 37763

7.0 General Relief Requests for Valves

General Relief Request:	VG-2
Valves:	Containment Isolation Valves
Category:	A, AC
Test Requirements:	Leak rate test in accordance with Subsection IWV-3421 through 3425 and IWV-3427(b).
Basis for Relief:	Containment isolation valves are required to be leakage rate tested in accordance with 10CFR50, Appendix J. The leakage rate requirement is based on a total allowable leakage rate for all valves instead of an individual valve leakage rate. IWV-2200(a) defines Category A as "valves for which seat leakage is limited to a specified maximum amount in the closed position of fulfillment of their function." Although, leakage rates for containment isolation valves are not limited on an individual basis, they have been determined to be Category A valves. Since containment isolation valves are Category A, the leakage rate testing requirements of IWV-3420 must be satisfied. The leakage rate testing performed per Appendix J satisfies the intent of IWV-3421 through 3425, however, it does not satisfy the individual valve leakage rate analysis and corrective actions of IWV-3426 and 3427. In order to prevent duplicate leakage testing of these valves, individual leakage rate will be obtained during Appendix J testing and the requirements of IWV-3426 and 3427(a) will be applied via separate procedure. The trending requirements of IWV-3427(b) does not provide meaningful results.
Alternate Testing:	Containment isolation valves will be leak rate tested in accordance with the 10CFR50 Appendix J testing program. In addition, individual valve leakage rates will be obtained by test or analysis and the requirements of IWV-3426 and 3427(a) will be applied via a separate procedure for those valves that are Appendix J Type C tested.

NIAGARA MOHAWK

TRANSMITTAL MEMO

813-018 N01-85 SYMBOL NO. 55-10-092

TO: Nuclear Regulatory Compliance - Gary Sanford

FROM: EL Ridings

SUBJECT: ASME - Leak Testing IAW IST-001 pursuant IWV-3420 DATE: 1-16-90

MESSAGE: General Relief Request VG-2 (Attached), clearly states containment isolation valves are required to be leak rate tested IAW 10CFR50 App J. TS Table 3.34 and FSAR Table VI-3b (Attached) identify primary containment Isolation Valves which are currently not being tested (IST-001 example pages attached). For those valves incorrectly labeled LA-R (Tested IAW ASME IWV-3420), the leakage rate specified by the owner is over 3000% higher than those permissible IAW IWV-3426 (see N1-ST-C19 attached). These leakage rates were incorrectly obtained from TS 4.2.7.1. IWV-3420 requires a 2 year frequency with no grace period. I recommend someone with LLRT experience and knowledge of 10CFR50 App A Criterion 55, 56, 57 review this matter immediately. This is, after all, what protects the public and demands top priority including re-start (see TS 4.3.3.e.1).

REPLY

DATE

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

d. Frequency

Three integrated leak rate tests shall be performed at approximately equal intervals during each 10-year service period with the third test in each ten-year interval corresponding with the ten-year scheduled in-service inspection shutdown.

e. Local Leak Rate Tests

(1) Primary containment testable penetrations and isolation valves shall be tested at a pressure of 35 psig each major refueling outage except bolted double-gasketed seals shall be tested whenever the seal is closed after being opened, and at least at each refueling outage.

(2) Personnel air lock door seals shall be tested once within 24 hours after opening when the reactor is in a power operating condition, at a pressure of 10 psig and the leak rate extrapolated to 35 psig. Air lock seals shall also be leak rate tested at a pressure of 35 psig at the beginning of each operating cycle. An additional 35 psig leak rate test shall be performed near the middle of the operating cycle should a shutdown requiring de-inerting arise. If the above shutdown does not occur or is not anticipated, the air lock seals will be

LIMITING CONDITION FOR OPERATION

3.3.3 LEAKAGE RATE

Applicability:

Applies to the allowable leakage rate of the primary containment system.

Objective:

To assure the capability of the containment in limiting radiation exposure to the public from exceeding values specified in 10 CFR 100 in the event of a loss-of-coolant accident accompanied by significant fuel cladding failure and hydrogen generation from a metal-water reaction.

Specification:

Whenever the reactor coolant system temperature is above 215 F the primary containment leakage rate shall be within the limits of 4.3.3.b.

SURVEILLANCE REQUIREMENT

4.3.3 LEAKAGE RATE

Applicability:

Applies to the primary containment system leakage rate.

Objective:

To verify that the leakage from the primary containment system is maintained within specified values.

Specification:

a. Integrated Primary Containment Leakage Rate Test

(1) Integrated leak rate tests shall be performed prior to initial Station operation at the test pressure of 35 psig (P_p) and the test pressure (P_t) of 22 psig to obtain the respective measured leak rates L_m (35) and L_m (22).

should be 30 ml/hr
or 6ml/hr for these 12" valves
or .0016 gal/min

10.0 ACCEPTANCE CRITERIA

10.1 Operation Review

10.1.1 Valve Test Results Meet ISI Limits

Valve ID	Corrected Leakage Rate (Step Number)	Acceptable ISI Leakage Rate	Valve Test Results
80-01	______ gpm (Step 8.1.24)	Less than or equal to 5 gpm	☐ Sat ☐ Unsat
80-21	______ gpm (Step 8.2.24)	Less than or equal to 5 gpm	☐ Sat ☐ Unsat
80-02	______ gpm (Step 8.3.24)	Less than or equal to 5 gpm	☐ Sat ☐ Unsat
80-22	______ gpm (Step 8.4.24)	Less than or equal to 5 gpm	☐ Sat ☐ Unsat

* If any valve leakage exceeds the acceptance limit or ISI acceptance criteria is exceeded, then the valve shall be declared immediately inoperable. Notify the ISI Department of the inoperable valve.

10.1.2 All test documentation is completed. ☐ YES ☐ NO

______________ / ______ / ______
Completed by ASSS/SSS Date Time

10.1.3 SSS Review

a. ☐ Satisfactory, no corrective action required.

b. ☐ Satisfactory, corrective action required (Use Remarks Section as necessary, and initiate a WR).

c. ☐ Unsatisfactory, (Use Remarks Section as necessary to explain, initiate a WR, and immediately NOTIFY* the Station Superintendent or Alternate).

Remarks:

____

____

______________ / ______ / ______
Signature, SSS Date Time

____
*Name of Person Notified

leakage shall be adjusted to function maximum pressure differential value. This adjustment shall be made by calculation appropriate to the test media and the ratio between test and function pressure differential, assuming leakage to be directly proportional to the pressure differential to the one-half power.

(f) Valves not qualifying for reduced pressure testing as defined in (e) above shall be leak tested at full maximum function pressure differential, with adjustment by calculation if needed to compensate for a difference between service and test media.

IWV-3424 Seat Leakage Measurement

Valve seat leakage may be determined by one of the following:

(a) draining the line, closing the valve, bringing one side to test pressure, and measuring leakage through a downstream telltale connection, or

(b) by measuring the feed rate required to maintain pressure between two valves or between two seats of a gate valve, provided the total apparent leak rate is charged to the valve or gate valve seat being tested, and that the conditions required by IWV-3423 are satisfied.

IWV-3425 Test Medium

The test medium shall be specified by the Owner.

IWV-3426 Analysis of Leakage Rates[2]

Leakage rate measurements shall be compared with previous measurements and with the permissible leakage rates specified by the plant Owner for a specific valve. If leakage rates are not specified by the Owner, the following rates shall be permissible:

(a) for water, at function pressure differential, 30*D* ml/hr;

(b) for air, at function pressure differential, 7.5*D* standard cu ft/day.

D is the nominal valve size, in.

IWV-3427 Corrective Action

(a) Valves with leakage rates exceeding either the values specified by the Owner, or those rates given in IWV-3426 shall be replaced or repaired.

(b) For valves 6 in. nominal pipe size and larger, if a leakage rate exceeds the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate by 50% or greater, the test frequency shall be doubled; the tests shall be scheduled to coincide with a cold shutdown until corrective action is taken, at which time the original test frequency shall be resumed. If tests show a leakage rate increasing with time, and a projection based on three or more tests indicates that the leakage rate of the next scheduled test will exceed the maximum permissible leakage rate by greater than 10%, the valve shall be replaced or repaired.

IWV-3500 INSERVICE TESTS, CATEGORY C VALVES

IWV-3510 SAFETY VALVE AND RELIEF VALVE TESTS

IWV-3511 Test Frequency

Valves shall be tested at the end of each time period as defined in Table IWV-3510-1.

IWV-3512 Test Procedure

Safety valve and relief valve set points shall be tested in accordance with ASME PTC 25.3-1976. Bench testing, with suitable hydraulic or pneumatic equipment, or testing in place with hydraulic or pneumatic assist equipment, is an acceptable method under PTC 25.3-1976. Valves so tested are not required to be additionally leak tested in accordance with IWV-3420.

IWV-3513 Additional Tests

When any valve in a system fails to function properly during a regular test, additional valves in the system shall be tested as determined by an arbitrary assumption that a 12 month operating period has passed to another refueling, and the additional valves shall be tested to make the cumulative total tested at least $N/60 \times$ total valves in this category, where *N* now includes the additional 12 months. (See Table IWV-3510-1 for definition of *N*.) If any of these additional valves fails to function properly on test, then all valves in the system in this category shall be tested.

IWV-3514 Corrective Action

A valve failing to function properly during test shall be repaired or replaced.

[2]For check valves, use double the listed values.

DWG. NO. C-18012-C SH.2

[VA]LVE NUMBER	CLASS AND DWG. COOR.	VALVE (CAT.)	SIZE (IN.) AND TYPE	ACTU. TYPE	NORM. POSIT.	TEST REQ.	STROKE DIRECT.	CHECK VALVE TEST DIRECT.	C.S JUST. OR RELIEF REQ. NO.	C.S OR ALT. TEST PERF.	REMARKS
80-21 [C]SP Pump [Isolat]ion Valve	2 H-7	A	12 GTV	MOA	O	FE-Q ST-Q PI-R LA-R ?	C				
80-22 [C]SP Pump [Isolat]ion Valve	2 H-9	A	12 GTV	MOA	O	FE-Q ST-Q PI-R LA-R ?	C				
80-15 []CTN-SP Inlet [D]rywell	2 D-3	B ?	12 GTV	APA	O	FE-Q ST-Q FS-Q PI-R	O,C O				NOTE 1
80-16 CTN-SP HT . Inlet IV	2 D-4	B	12 GTV	APA	O	FE-Q ST-Q FS-Q PI-R	O,C O				NOTE 1
80-17 []CTN-SP Loop [Chec]k Valve to [Dryw]ell Sparger	2 F-3	C	12 CHV	SEA	DE	FE-Q		F	RR-1	PE-R	
80-18 CTN-SP Loop [Chec]k Valve to [Dryw]ell Sparger	2 F-4	C	12 CHV	SEA	DE	FE-Q		F	RR-1	PE-R	
80-19 []2 CTN-SP Loop [Chec]k Valve to	2 I-4	C	3 CHV	SEA	DE	FE-Q		F	RR-1	PE-R	

? Containment Iso Valve!

NO Leak Test

IST-001-Rev 2

LIMITING CONDITION FOR OPERATION

Table 3.3.4 Continued

PRIMARY CONTAINMENT ISOLATION VALVES

LINES ENTERING FREE SPACE OF THE CONTAINMENT

Line or System	No. of Valves (Each Line)	Location Relative to Primary Containment	Normal Position	Motive Power	Maximum Oper. Time (Sec)	Action on Initiating Signal	Initiating Signal (All Valves Have Remote Manual Backup)
Core Spray (c)							
Pump Suction (Four Lines from Suppression Chamber) 81-11/12/21/22	1	Outside	Open	AC Motor	90	-	Remote manual
Pump Discharge (Two Test Lines to Suppression Chamber) 40-05/06	1	Outside	Closed	AC Motor	90	Close	Reactor water level low-low
LINES WITH A CLOSED LOOP INSIDE CONTAINMENT VESSELS							
Recir. Pump Cooling Water Supply (c)							
Supply Line 70-93	1	Outside	Open	Self Act. Ck.	--	-	-
Return Line 70-92	1	Outside	Open	DC Motor	30	-	Remote manual
Drywell Cooler Water Supply (c)							
Supply Line 70-95	1	Outside	Open	Self Act. Ck.	--	-	-
Return Line 70-94	1	Outside	Open	DC Motor	30	-	Remote manual
LINES WITH A CLOSED LOOP OUTSIDE CONTAINMENT VESSELS							
Containment Spray (c)							
Drywell & Suppression Chamber Common Supply (Four Lines) 80-15/16/35/36	1	Outside	Open	Air/DC Sol.	60	Open	Reactor level low-low and high drywell pressure
Drywell Branch (Four Lines) 80-17/18/37/38	1	Outside	-	Self Act. Ck.	--	-	-
Suppression Chamber Branch (One Branch for Each System) 80-65/66/67/68	2*	Outside	-	Self Act. Ck.	--	-	-
Pump Suction from Suppression Chamber (Four Lines) 80-01/02/21/22	1	Outside	Open	AC Motor	70	-	Remote manual

*One valve in each separate line and one valve in each common line.

(c) These are classified as not-testable valves and penetrations.

NC LLRT Performed

TABLE VI-3b (Continued)

PRIMARY CONTAINMENT ISOLATION AND BLOCKING VALVES
LINES ENTERING FREE SPACE OF THE CONTAINMENT

Line or System	No. of Valves (Each Line)	Location Relative to Primary Containment	Normal Position	Fail Position on Loss of Motive Power or Control Signal	Motive Power	Maximum Oper. Time (Sec)	Action on Initiating Signal	Initiating Signal (All Valves Have Remote Manual Backup)
Reactor Cleanup System Relief Valve Discharge Line Vacuum Relief (One Line) [illegible]	2	Outside	-	-	Self. Act. Ck.	--	-	
Post Accident Reactor Sampling Return Line (One Line) 63-05/04	2	Outside (a)	Closed	Closed	Pn/D.C. Solenoid	30	Close	Reactor water level low-low or drywell high pressure
Core Spray								
Pump Suction (Four Lines From Suppression Chamber) 81-1/2/21/22	1	Outside	Open	As Is	A.C. Motor	90	-	Remote manual
Pump Discharge (Two Lines to Reactor Vessel) 40-02/12 40-01/09 40-10/11	1	Outside	Open	As Is	A.C. Motor	-	Open	Locked open
	2(e)	Inside	Closed	As Is	A.C. Motor	-	Open	Low-low reactor pressure
Pump Discharge (Two Test Lines to Suppression Chamber) 40-05/06	1	Outside	Closed	As Is	A.C. Motor (d)	90	Close	
Raw Water Intertie to Core Spray (Two Lines) 43-71/74	1	Outside	Closed	As Is	A.C. Motor	35	-	-
43-53/54	1	Outside	Closed	-	Self Act. Ck.	--		
H_2-O_2 #11 Sampling								
Drywell Supply (Two Lines) 201.7-1/2-3/4	2	Outside(b)	Open	Closed	Pn/D.C. Solenoid	60	Close	Reactor water level low-low or high drywell pressure
Suppression Chamber Supply (One Line) 201.2-110/111	2	Outside(b)	Open	Closed	Pn/D.C. Solenoid	60	Close	
Drywell Return(c) (One Line) 201.7-10/11	2	Outside (b)	Open	Closed	Pn/D.C. Solenoid	60	Close	
Suppression Chamber Return (One Line) 201.2-112/109	2	Outside (b)	Open	Closed	Pn/D.C. Solenoid	60	Close	

Rev. 2

TABLE VI-3b (Continued)

PRIMARY CONTAINMENT ISOLATION AND BLOCKING VALVES
LINES ENTERING FREE SPACE OF THE CONTAINMENT

Line or System	No. of Valves (Each Line)	Location Relative to Primary Containment	Normal Position	Fail Position on Loss of Motive Power or Control Signal	Motive Power	Maximum Oper. Time (Sec)	Action on Initiating Signal	Initiating Signal (All Valves Have Remote Manual Backup)
Containment Spray								
Drywell & Suppression Chamber Common Supply (Four Lines) 80-15/16/35/36	1	Outside	Open	Open	Pn/D.C. Sol.	60	-	Remote Manual Rev. 4
Drywell Branch (Four Lines) 80-17/18/37/38	1	Outside	-	-	Self Act. Ck.	—	-	-
Suppression Chamber Branch (One Branch for Each System) 80-65/66/67/68	2(f)	Outside	-	-	Self Act. Ck.	—	-	-
Pump Suction From Suppression Chamber (c) (Four Lines) 80-01/2/21/22	1	Outside	Open	As Is	A.C. Motor	70	-	Remote Manual
Containment Spray to Waste Bldg. (One Line) 80-114 80-115	2	Outside	Closed	As Is	A.C./D.C. Motor	90	Close	Reactor level low-low or high drywell pressure

NO LLRT PERFORMED

— [illegible] for LLRT test

Notes:

Pn - Pneumatically Operated

(a) These valves may be open for containment inerting, deinerting, sampling or N_2 makeup.
(b) These valves have the capability of being opened after a closure signal. This allows for H_2-O_2 monitoring and N_2 purge.
(c) This is also the Containment Atmosphere Monitoring line.
(d) These valves have their motive power removed during normal station operation.
(e) These valves are parallel.
(f) One valve in each separate line and one valve in each common line.

Rev. 4

This requirement affects one respondent and, therefore, is not subject to Office of Management and Budget review under P.L. 96-511.

Sincerely,

Original Signed By:

Donald S. Brinkman, Senior Project Manger
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosure:
Information Requested
By Office of Nuclear
Reactor Regulation

cc w/enclosure:
See next page

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no legal objection

PDI-1:LA	PDI-1:PM DSB	OGC	PDI-1:D		
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