William J. Cahill, Jr.
Vice President

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4 Irving Place, New York, N Y 10003
Telephone (212) 460-3819

November 17, 1975

RE: Indian Point Unit No. 3
Docket No. 50-286

Mr. R. C. De Young
Assistant Director for
Light Water Reactors, Group 1
Division of Reactor Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. De Young:

On November 12, 1975 we forwarded to you responses to the majority of outstanding items listed in your letter of November 5, 1975. The purpose of the present submittal is to provide responses to those remaining items not covered in our November 12 letter; namely items 4, 5c, and 6. Please find these responses in the Attachment.

In addition, certain information has been requested by members of your staff. This information follows:

1. The Loose Parts Monitor referred to on page 5 of Supplement No. 1 to the Safety Evaluation Report by the Directorate of Licensing dated January 16, 1975 has been installed.

2. Monitors as described in the response to FSAR Question 11.14 have been installed on the blowdown flash tank vents of Indian Point Units 1 and 3.

3. The closure times of all safety injection valves have been measured in accordance with Test Procedure 4.10.1. Some valves were found to have stroke times exceeding the nominal 10-second limit prescribed in the test procedure. These particular valves and their respective measured stroke times were reviewed and evaluated by both the Joint Test Group and by Westinghouse and were found to be acceptable.

13144

TO: R. C. DeYoung November 17, 1975

RE: Indian Point Unit No. 3
Docket No. 50-286

4. A preoperational test of the air locks at peak accident pressure has been performed in accordance with Appendix J to 10 CFR 50. The measured leakage rate for the air locks was added to the leakage rates obtained for the other Type B and Type C preoperational tests. The combined leakage rate of all penetrations and valves subject to Type B and Type C tests was less than 0.6 La and was therefore in accordance with the requirements of Appendix J.

5. Attached is a copy of the procedure for the Service Water Functional Test, including the test results.

6. Also attached is a supplement to our response of November 12, 1975 to Item H.5, regarding the sample line penetration.

Very truly yours,

William J. Cahill, Jr.
William J. Cahill, Jr.
Vice President

Sworn to before me this
17th day of November, 1975.

Donal F. McCarthy
Notary Public

DONAL F. McCARTHY
NOTARY PUBLIC, State of New York
No. 30-2601325
Qualified in Nassau County
Commission Expires March 30, 1977

Copy to: James P. O'Reilly, Director
Office of Inspection and Enforcement
Region 1
U.S. Nuclear Regulatory Commission
King of Prussia, Penn. 19406

ATTACHMENT

Responses to Enclosure 2 to
November 5, 1975 NRC Letter

Item 4 Reactor Pressure Vessel Supports

In this item you request further information on the adequacy of the supports for the Indian Point Unit #3 reactor vessel when the effects of asymmetric pressure distributions on the vessel and internals are considered.

We have performed preliminary analyses on Indian Point #3 of the vessel, internals and the interaction with the reactor coolant loop for a postulated pipe break at the reactor vessel inlet nozzle. Included in the analyses were the effect of the cavity pressurization, the internals depressurization effects and the loop reaction loads at the attached nozzles. This preliminary analysis included the following points:

1) The break opening area used in these evaluations was calculated from clearances in the pipe annulus and expected motion of the reactor vessel and pump. This break area was 500 in^2.

2) The stiffness of the reactor vessel support and the shield wall concrete below the support was included in the evaluation. The stiffness of the reactor vessel support structure used in the analysis is 23×10^6 lb/in.

3) The cavity pressure was evaluated using current plant flow blockage assumptions. The concrete design both in the reactor cavity and the piping annulus is such that the cavity pressurization force is the predominant load of the applied loads. The magnitude of the horizontal cavity load applied to the vessel is approximately 9600 kips.

4) The combined load applied to the vessel from the internals and loop reaction forces were ascertained to be approximately 6000 kips. This was determined from analyses on similar four loop plants both with and without fluid-structure interaction. The load noted is typical of that expected if a detailed individual plant blowdown analysis was performed for Indian Point #3 including the effect of fluid-structure interaction.

5) If these input forces were summed on a time history basis and were dynamically applied to a structural model of the reactor vessel system, the resulting maximum dynamic load induced in the support system would be approximately 16,000 kips.

6) The reactor vessel supports will reach the faulted condition limit of 1.2 S_y at approximately 3600 kips. The reactor coolant loop supports will reach their faulted condition limits at a load magnitude induced in the loops of 6500 kips. This indicates that the total resisting load the vessel supports and loop supports can develop is at least 10,000 kips. Greater capacity for applied load could be developed if elastic-plastic analyses were performed.

We have concluded from this preliminary analysis, which satisfied the intent of Item 3 of your July 22, 1975 letter, that detailed inelastic system analyses would be required to realistically assess the effect of a postulated break at the vessel inlet nozzle.

Analyses and analytical experiences in similar plants have also allowed us to evaluate the variation of the load with respect to break opening area. Based on such analyses, a break opening of approximately 70 to 80 square inches will produce a load of about 4500 kips which can be resisted by the system supports within their elastic limits. A break area of about 140 to 150 square inches will produce a load of about 10,000 kips, which is equivalent to the restraint capability established by preliminary analysis as stated in paragraph 6, above. This fact, coupled with the fact that the system was shown adequate to the requirements of the FSAR for breaks postulated outside the shield wall in the hot and cold leg and for breaks in the crossover leg, demonstrates that there is certainly a large spectrum of breaks for which the system is adequate elastically. Only for breaks of opening area greater than 140 to 150 square inches at the vessel nozzles would detailed inelastic system analysis be required to verify the system capability. The inelastic system analysis was requested in Item 4 of your July 22, 1975 letter.

Complex and advanced methods are required to perform an inelastic analysis on a system as large and complex as the reactor coolant system for Indian Point #3. All components, piping and component supports, including those of the pump and steam generator, must be included in the analysis with the proper representation of their elastic and inelastic behavior. This representation is particularly difficult for the support structures of the reactor coolant pump and steam generator, which are highly redundant frame structures with many members. An analysis which includes these features on a system this large has not been performed to date at Westinghouse, and, to our knowledge, has not been performed in the industry. We have estimated that over a year of development would be necessary to generate the methods and analytical techniques required to solve this problem.

If such an inelastic analysis were performed, we feel the system capability would be verified. The system has a high potential for resisting loads above the yield strength of the material due to its strain hardening characteristics and would therefore retain its structural configuration. Ruptures in other loops would not be expected and the core should retain a coolable geometry. These events would assure that cooling water will be supplied to a coolable core and that plant safety will not be violated.

Furthermore, we have reviewed the ECCS piping and feel that even with the postulated limited reactor vessel displacement, the integrity of the piping would be maintained so that the emergency core cooling system could provide cooling water to the reactor.

Because of the complexity and difficulty of the inelastic analysis, we chose to propose in our September 30, 1975 letter an augmented program of inservice inspection to assure safety to the plant for the limited spectrum of breaks which analytically exceeded the elastic limit of the supports. This inspection will assure that there are no flaws in the piping in the inspection region which could propagate and cause a loss-of-coolant-accident.

We have considered the requests of Items 3 and 4 of your July 22, 1975 letter. Item 3 was specifically evaluated in the preliminary analyses performed and discussed in this letter. Item 3 demonstrated that inelastic analysis was necessary. We examined in detail the requirements for an inelastic analysis to the requirements of Item 4 and determined that we could not provide the results requested in a timely fashion and, as an alternative chose to assure the safety of the plant by lowering the probability of the break in question to an extremely small value by augmenting the inservice inspection.

Our preliminary analysis indicates that the plant is safe for a large spectrum of breaks including limited area breaks at the vessel nozzles. Only for very large area breaks at the vessel nozzles would detailed inelastic analyses be required to analytically verify the design adequacy. Augmented inservice inspection provides and assures a safe margin in the plant for these breaks.

Item 5 - Conformance with Appendix J to 10CFR50

(C) Isolation Valves Not Included in Table 4.4-1

Table 4.4-1 of the proposed Tech. Specs. has been prepared listing those containment isolation valves that are to receive periodic tests in accordance with Appendix J to 10CFR50. Other valves provided to meet containment isolation requirements of Section 5.2 in the IP-3 FSAR are listed below with supporting justification that they not be included in Table 4.4-1.

Valve	Function	Discussion
822A (MOV) 822B (MOV) 751A(Check V) 751B(Check V)	Component Cooling System Cooling water to residual Heat Exchangers	These valves are closed during plant operation and open during accident conditions (MOV 822 A & B receive an open signal). The portion of the CCS piping inside containment is a closed system: that portion of the component cooling system outside containment supplying cooling water to the residual heat exchangers is also a closed system and monitored for radioactivity. As these four valves are open and in use during the post-accident condition, and as this cooling system is a closed and monitored water sealed system both inside and outside containment, a Type C test, as required by Appendix J, would serve no particular purpose. These

Valve	Function	Discussion
		valves are noted as containment isolation valves in accordance with Section 5.2 of the FSAR. For these valves, the intent of GDC-54 and GDC-57 has been met. GDC-55 and GDC-56 do not apply.
752F 753F 752 J 753 J	Component Cooling System cooling water to Recirculation Pumps	During both the normal plant operation and post-accident condition, these manually operated valves are in the open position to provide cooling water to the Recirculation pumps located within containment. The portion of this cooling system within containment is a closed system, that portion of the CCS outside containment supplying cooling wate to these pumps is also a closed system and monitored for radioactivity. As these four valves are open and in use during plant operation and the post-accident condition and as this cooling system is a closed and monitored water sealed system both inside and outside containment, a Type C test, as required by Appendix J, would serve no particular purpose. These valves are noted as containment isolation valves in accordance with Section 5.2 of the FSAR. For these valves, the intent of GDC-54 and GDC-57 has been met. GDC-55 and GDC-54 do not apply.

Valve	Function	Discussion
PCV-1111 PCV-1111	Air Supply to the Weld Channel and Penetration Pressure System	During plant operation, as well as post-accident conditions, these isolation valves are required to be in the open position to assure continuous pressurization of the Weld Channel and Penetration Pressurization System. As the supply air to PCV-1111 is always higher than peak accident pressure within containment, no potential exists for leakage from the containment through these valves. Therefor, a Type C test for Valves PCV-1111 would serve no purpose. These valves are noted as containment isolation valves in accordance with Section 5.2 of the FSAR: the Weld Channel and Penetration Pressurization System within containment is considered the closed system. Thus, the intent of GDC-54 and GDC-57 has been met. GDC-55 and CDG-56 do not apply.
CB-1 (2)	Pressure Relief of Air Lock Compartment	Check Valves CB-1 are included in the design of the containment air lock to relieve any pressure buildup between the air lock doors, should air from the WC & PPS leak into the space between the closed air lock doors. This feature of venting the annulus between the air lock doors is provided as protection

Valve	Function	Discussion
		to operating plant personnel for those instances when either of the doors is to be opened; however, for the instance of a postulated LOCA occurring when the inner door is closed and the outer door is open, check valves CB-1 become a part of containment isolation boundary and, therefore, are subject to Type C tests required by Appendix J to 10CFR Part 50. Valves CB-1 will be included in Table 4.4-1 of the Tech. Specs.

The IP-3 containment isolation system is discussed in the FSAR, Section 5.2. This section includes a discussion of the criteria utilized in the design of all containment penetrations, including steam and feedwater lines to the steam generators. This criteria was included in the PSAR as part of the construction permit application, as well as the FSAR in connection with the application for the operating license. Relative to the steam and feedwater lines and their connection to the steam generators, these lines are not classified in Section 5.2 of the FSAR as containment penetrations. Rather, the lines and shell side of the steam generator are considered an extension of containment boundary. This carries with it the requirement that steam generator shell, feed lines and steam lines within containment be designed to meet seismic and missile protection criteria. The fact that the valves in the steam lines and feed water lines are not designated as containment isolation valves is appropriate for the following reasons:

1. The ASME Code requires that a self actuated safety valve be provided on the steam line for overpressure protection of the steam generator. Should the main steam isolation valve be designated for containment isolation purposes, these self actuated safety valves would thus become part of the containment isolation boundary for the containment penetration. However, for the instance of a failed or leaking valve no provisions can be made for valving off of the self actuated safety valve. Thus these valves are precluded from serving as a containment isolation feature.

2. Power operated relief valves, installed in parallel to self actuated safety valves discussed above, are included as part of the reactor control system as well as for overpressure protection of the steam generator. Should the main steam isolation valve be designated for containment isolation purposes, the power operated relief valves would then become part of the isolation boundary for this containment penetration. Because these valves are provided for system design requirements for the RCS and secondary system they can not be included in the scope of requirements for containment isolation.

3. Requirements for operation of the steam driven auxiliary feed pump dictate that steam be supplied to this pump during post-accident conditions. The remote operated valve controlling this steam supply would similarly become part of containment isolation. To designate such a valve as a containment isolation valve would be meaningless when it is intentionally open during the post-accident condition.

4. As the auxiliary feed pumps are started automatically during post-accident conditions to maintain a minimum water level within the steam generators, system requirements are therefore established that valves in the feedwater line be open. This auxiliary feedwater system pressure however, would be higher than accident pressure within the containment, so that no potential leakage would occur through that path.

5. As part of the Type A test required by Appendix J, the steam and feed lines and steam generator shell serve as that portion of containment boundary, thus assuring that no unanticipated leakage would go undetected on that area of containment boundary.

6. Consideration of radioactivity leakage to the outside environment from within containment through the steam line or feed line penetrations has been given consideration elsewhere in the plant design and operating requirements. Specifically, limits are placed in the Tech. Specs. on secondary side activity, primary to secondary leakage rate, and steam generator tube integrity. These operating limits give appropriate consideration for the instances of, site boundary dose calculations resulting from a steam or feed line break outside containment or malfunction of the power-operated relief valves or self-actuated safety valves.

Thus, to designate valves in the steam and feed lines as containment isolation valves is neither consistent nor compatible with other system requirements that would dictate use of these lines during the post-accident condition. Based on this justification, these lines are not considered as containment penetrations and, therefore, do not carry valves designated for containment isolation purposes subject to Appendix J testing.

Item 6 - Containment Isolation

The following group of valve listing comprise all the non-automatic containment isolation valves which we intend to have open, either continuously or intermittently, during power operation or at other times when containment integrity is required.

The four groups of C.I. valve listing are:

1) Valves Required to be Open during a Postulated Accident - These valves will be open during normal power operation as well as during and/or after the postulated accident. For example, these valves are located in piping of protection, safeguards or essential service systems which will be required to function during and/or after the postulated accident.

2) Valves Normally or Intermittenly Open During Normal Plant Operation - These valves will be open during normal plant operation and are located within piping systems which either service equipment within the containment or are only intermittently opened for periodic testing of safeguards systems, or for calibration purposes. All valves in this category will be manually closed if the postulated accident should occur.

3) Valves Normally Closed for Normal Operation but can be Administratively Opened During Normal Operation - These valves will be normally closed during normal power operations and for postulated accident conditions. However should the need arise for any of these valves to be opened during normal plant operations, they will be administratively opened and administratively closed as required. The probability of a valve in this category being opened and a simultaneous LOCA occuring is extremely small due to the infrequent amount of time the valve is opened.

Even though the probability of a simultaneous LOCA and a valve in this category being opened is extremely small, should this event ever occur, then the valve would be manually closed during the post LOCA period.

3a) Valves Normally Closed for Normal Operation but can be Administratively Opened During Normal Operation and are Required to be Opened Intermittently Following a Postulated Accident - These valves will be normally closed during normal power operations and for postulated accident conditions. However, should the need arise for any of these valves to be opened during normal plant operations, they will be adminstratively opened and administratively closed as required. In addition, these valves are required to be opened intermittently following a postulated accident. For instance, valves in post-accident sampling systems are in category 3a).

The following valves are category 1) - <u>Valves Required to be Open During a Postulated Accident</u>

Valve	Function	Discussion
744	Low Head S.I. to R.C.S.	This valve is open and de-powered per Tech.Spec. requirements to preclude spurious valve closure during post accident period. Valving on this penetration complies with the intent of GDC 56. GDC 55 and GDC 57 do not apply.
869A 869B	Spray lines(2)	Each of these valves is a double disc gate valve with seal water injected between the discs, as described in the IP-3 FSAR, Section 5.2. This feature meets the intent of the double valving described in GDC 56 for containment isolation. These valves are locked open during operation and are open during the post accident period. No auto closures are provided to preclude spurious valve closure and interruption of spray system flow. GDC 55 and GDC 57 do not apply.
850A 851A	Hi-Head S.I. Line to R.C.S.	These valves are open during plant operation, as well as during the post accident period. No auto closures

Valve	Function	Discussion
850A 851A(Cont'd)		are provided to preclude spurious valve closure and interruption of S.I. System flow. These valves meet the intent of double valving for containment penetration isolation described in GDC 56. GDC 55 and GDC 57 do not apply.
752F 752J 753F 753J	Cooling Water to Recirculation Pumps (2)	These valves are open during the post accident period, as well as during plant operation. Auto closures are not provided to preclude spurious valve closure and interruption of cooling water to the recirculation pumps, These valves, along with the closed motor cooling system within containment, meet the intent of GDC 57 for isolation of the containment penetrations. GDC 55 and GDC 56 do not apply.
SWN-41 SWN-44 SWN-51 SWN-71 SWN-41 SWN-44 SWN-51 SWN-71 SWN-41 SWN-44	S.W. Supply to Containment Fan & Motor Coolers(5)	These valves are open both during plant operation and the post accident period. Auto closures are not provided to preclude spurious valve closure and subsequent loss of cooling water to the FC's. These valves together with the closed fan cooler system inside containment meet the

Valve	function	Discussion
SWN-51 SWN-71		intent of GDC 57 for isolation of these penetrations. GDC 55 and GDC 56 do not apply.
SWN-41 SWN-44 SWN-51 SWN-71		
SWN-41 SWN-44 SWN-51 SWN-71		
PCV-1111 PCV-1111	Air Supply Lines (2) to WC & PPS	These valves are open during plant operation, as well as during the post-LOCA period. Not auto-closures are provided to preclude spurious valve actuation and interrupt the air supply to the WC & PPS. These valves, together with the closed WC &PPS inside containment, meet the intent of GDC 57 for isolation of these containment penetrations. GDC 55 and GDC 56 do not apply.
1814A 1814B 1814C	Lines to Containment pressure transmitters	These valves are open during plant operation, as well as during the post accident. Auto-closures are not provided to preclude spurious closure and thus interrupt status of containment pressure. These valves meet the intent of GDC

Valve	Function	Discussion
1814A(Cont'd) 1814B(Cont'd) 1814C(Cont'd)		56 for instrument lines. GDC 55 and GDC 57 do not apply.

The following valves are category 2) - Valves Normally or Intermittently Open During Normal Plant Operation

Valve	Function	Discussion
550	N_2 Supply to P.R.T.	Valve 550 furnishes one of two valving provisions for isolation of this penetration. This isolation valve is open for proper system performance. No automatic closure of 550 is provided to preclude spurious valve closure and interruption of N_2 supply. Valve closure is made manually during post accident period. As Valve 550 and Valve 518 provide double means to establish containment isolation of this penetration (of which one of the valves is an automatic closure, i.e., check valve 518), the intent of GDC 56 has been met. (GDC 55 and GDC 57 do not apply.
1870 743	RHR Mini-Flow Line	These valves furnish two isolation provisions for this containment penetration. No auto closures are provided on this line to assure that the RHR pump will not suffer damage due to a spurious S.I. signal or pump start. Valve closure is made manually during post accident period. As two valves are provided in this penetration

Valve	Function	Discussion
1870 (cont'd.) 843		for containment isolation purposes, the intent of GDC 56 has been met. GDC 55 and GDC 57 do not apply.
205 226 227	Charging Line to R.C.S.	These valves furnished two isolation provisions for this containment penetration. No auto closures are provided to preclude damage to the charging pumps due to spurious S.I. signal or valve closure. Valve closure is made manually during the post accident period. As two valves are provided in this penetration for containment isolation purposes, the intent of GDC 56 has been met. GDC 55 and GDC 57 do not apply.
250A 241A 250B 241B 250C 241C 250D 241D	Seal Water Supply Line to R. C. Pumps (4)	These valves furnish two isolation provisions for each of these containment penetrations. No auto closures are provided to preclude damage to the R.C. pump seal due to spurious S. I. signal or valve closure. Valves are closed manually during post accident period. As two valves are provided in each penetration for containment isolation purposes, the intent of GDC 56 has been met. GDC 55 and GDC 57 do not apply.

Valve	Function	Discussion
878A 878B	Spray Pump Test Line	These valves comply with the intent of GDC 56 for containment isolation purposes. GDC 55 and GDC 57 do not apply. These valves are closed during the post accident period and only opened for periodic spray pump testing.
859A 859C	Hi-Head Pump Test Line	These valves are closed during the post accident period and only opened during plant operations for periodic S. I. pump testing. These valves meet the intent of double valving for containment isolation described in GDC 56. GDC 55 and GDC 57 do not apply.
1833A 1833B	BIT Bypass Line	
1610	N_2 Supply to R.C.D.T.	This valve furnishes one of two valving provisions for containment isolation of this penetration. This valve is open to provide for proper system performance. No automatic closure of 1610 is provided to preclude spurious valve closure and interruption of N_2 supply. Valve closure is made manually during the post accident period. As Valve 1610 and Valve 1616 provide the double valving means to establish containment isolation on this penetration, of which one of the valves (1616) is an automatic closure (i.e., check valve), the intent of GDC 56 has been met. GDC 55 and GDC 57 do not apply.

Valve	Function	Discussion
891A 891B 891C 891D 863	N_2 Supply to Accumulators	These valves are opened intermittently during plant operation but are closed manually during the post accident period. Auto closures are not provided to preclude spurious valve opening or closing and interruption of the system performance. These valves provide the double valving means to establish containment isolation on the penetration and the intent of GDC 56 has been met. GDC 55 and GDC 57 do not apply.
580A 580B	Pressurizer Pressure Transmitter Calibration	These valves are closed during the post accident period and only opened periodically during plant operation for calibration purposes. The valves meet the intent of GDC 56 for double valving of containment penetrations. GDC 55 and GDC 57 do not apply.
732	RHR System suction line from RCS	Open intermittently during periods when RHR System is in operation. Normally closed during power operation. This valving meets the intent of GDC 56. GDC 55 and GDC 57 do not apply.
958 959 990C	RHR System Sampling line	Used intermittently for sampling purposes when RHR System is in operation. Normally closed during power operation. This valving meets the intent of GDC 56. GDC 55 and GDC 57 do not apply.

Valve	Function	Discussion
MOV 885A, B	Containment Sump Recirculation Line	Valves are normally closed but may be opened during post LOCA if the normal recirculation path from the recirculation pump is not available. The two motor operated double disc gate valves located in series to provide the double barrier described in GDC 56. GDC 55 and GDC 57 do not apply.

The following valves are Category 3 - Valves Normally Closed for Normal Operation but can be Administratively Opened During Normal Operation

VALVE	FUNCTION	DISCUSSION
UH - 37 UH - 38	Auxiliary Steam to Containment	These valves are locked closed for normal plant operations, and the discussion of category 3) valves, as described above, applies to these valves. Each of these valves is a double disc gate valve with seal water injected between the discs, as described in the IP - 3 FSAR, Section 5-2. This feature meets the intent of the double valving described in GDC 56. GDC 55 and GDC 57 do not apply.
SA - 24 SA - 24	Service Air to Containment (1 penetration, 2 valves)	These valves are normally closed for normal plant operations, and the discussion of category 3) valves, as described above, applies to these valves. These valves are located in series and furnish two isolation provisions, and the intent of GDC 56 has been met. GDC 55 and GDC 57 do not apply.

The following valves are Category 3a) - Valves Normally Closed for Normal Operation but can be Administratively Opened During Normal Operation and are Required to be Opened Intermittently following a Postulated Accident - The discussion of Category 3a) valves, as described above, applies to all valves listed below.

1882 A	O_2 to Containment	Valve 1882 A furnishes one of two valving provisions for isolation of this penetration. This isolation valve is intermittently opened during the post LOCA period for proper system performance of the H_2 recombiner. Valve 1882 A combined with automatic isolation valves IV - 2A and IV - 2B provide the double means to establish containment isolation of this penetration, and the intent of GDC 56 has been met. GDC 55 and GDC 57 do not apply.
1875 A,B 1876 A,B	H_2 to Recombiner (4 penetrations)	Each of these valves furnishes one of two valving provisions for isolation of the penetrations in which they are located. The isolation valves are intermittently opened during the post LOCA period for proper recombiner operation. These valves combined with automatic isolation valves IV - 3A, 3B, 5A, 5B provide double

VALVE	FUNCTION	DISCUSSION
1875 A,B 1876 A,B	H_2 to Recombiner (Cont'd) (4 penetrations)	means to establish containment isolation of each penetration, and the intent of GDC 56 has been met. GDC 55 and GDC 57 do not apply.
PS 7, 8, 9, 10	Post Accident ventilation system Exhaust line	These valves may be opened intermittently during post accident venting. Valve PS-7 is the first of two valving provisions for containment isolation. Valves PS 8, 9, and 10 provide for the second of two valving provisions in the branch lines that tee from the header in which valve PS-7 is located. As double means are provided to establish containment isolation for this penetration, the intent of GDC 56 has been met. GDC 55 and GDC 57 do not apply.
MOV 888 A, B	RHR to SIS	Each of these motor operated valves, located in paralled branch lines, is a double disc gate valve with nitrogen injected between the discs. This feature meets the intent of double valving described in GDC 56. GDC 55 and GDC 57 do not apply.
1890 A, B, C, D, E, F, G, H, J	Post Accident Containment Sampling	Valves are opened intermittently during post accident for sampling. Double barriers are established in all headers and branch connections, and thus the intent of GDC 56 is met. GDC 55 and GDC 57 do not apply.
990 A, B	Recirculation Pump Discharge Sample Line.	These valves, which are located in series, provide the double barrier described in GDC 56. GDC 55 and GDC 57 do not apply.

Attached is the Standard Operating Procedure "SOP-CB- Revision 1, Containment Integrity", as well as the applicable portions of emergency procedure "PEP-ES-1A, Revision 1, "Loss of Coolant to Containment".

INDIAN POINT STATION

UNIT NO. 3

SOP-CB-1 REV. 1

CONTAINMENT INTEGRITY

SOP-CB-1

Containment Integrity

1.0 Intent

To provide a procedure for ensuring that the requirements for containment integrity are in effect.

2.0 Precautions and Limitations

2.1 As stated in Technical Specifications containment integrity is defined to exist when:

A) All non-automatic containment isolation valves which are not required to be open during accident conditions except those required to be open during normal plant operation are closed and blind flanges are installed where required.

B) The equipment door is properly closed.

C) At least one door in each personnel air lock is properly closed.

D) All automatic containment isolation valves are either operable or in the closed position, or isolated by a closed manual valve or flange that meets the same design criteria as the isolation valve.

E) The containment leakage meets the requirements of the integrated leak rate tests.

2.2 In accordance with Technical Specifications, Containment Integrity shall be maintained at all times except when:

A) The reactor is in the cold shutdown condition and the reactor vessel head is bolted in place, or

B) The reactor is in the cold shutdown condition with the shutdown margin being 10% $\Delta K/K$ if the reactor vessel head is completely unbolted.

3.0 Initial Conditions

None.

4.0 Instructions

Verify or establish the following conditions:

4.1 The required non-automatic containment isolation valves, listed below, are closed.

Fuel Transfer Canal Gate Valve		______	
High Head SI Pump Test Line	859A	______	
High Head SI Pump Test Line	859C	______	
Aux.Steam Supply to Containment(Locked Closed)	UH-37	______	
Aux.Steam Condensate Return(Locked Closed)	UH-38	______	
Recirculation Pump Sample Line	990A	______	
Recirculation Pump Sample Line	990B	______	
Station Air Supply to Containment	SA-24	______	
Station Air Supply to Containment	SA-24	______	
Post Accident Containment Sampling from 33 FCV	1890D, 1890B	______	
Post Accident Containment Sampling from 34 FCV	1890E	______	
Post Accident Containment Sampling from 31 FCV	1890A	______	
Post Accident Containment Sampling from 32 FCV	1890C	______	
Post Accident Containment Sampling from 35 FCV	1890F	______	
Post Accident Containment Sampling Return A	1890G	______	
Post Accident Containment Sampling Return B	1890H, 1890J	______	
RHR Suction (Locked Closed)	732	______	
Vent Between 730 and 732	A-106	______	1
RHR Sample	958	______	
RHR Sample	959	______	
RHR Sample	990C	______	
Vent Between 869A and 867A and 869B and 867B		______	
RHR Suction from Containment Sump	MOV-885A	______	
RHR Suction from Containment Sump	MOV-885B	______	
RHR Supply to High Head Pumps	MOV-888A	______	
RHR Supply to High Head Pumps	MOV-888B	______	
Containment Spray to RWST (Locked Closed)	878 A, B	______	
BIT Bypass	1833 A, B	______	1
Accumulator N_2 Supply	891 A,B,C,D	______	
Accumulator N_2 Supply	863	______	
Dead Weight Calibrator	580A/580B	______	
O_2 Supply to Containment	1882A	______	
H_2 Supply to H_2 Recombiner	1875A/1875B 1876A/1876B	______	
PACVS Exhaust Line	PS-7, PS-10, PS-8 PS-9	______	

With the exception of the fuel transfer canal gate valve, the valves listed above may be opened if necessary for operation, testing, etc., with the approval of the Watch Foreman. They should remain open only as long as necessary to perform the required evolution and an operator must be notified to immediately close the valve if required.

4.2 All automatic containment isolation valves listed below are operable or closed. Valves may be considered operable if the last required safety injection system test has been satisfactorily performed and the inoperable equipment list shows no valve inoperable.

SJAE to Containment	PCV-1229/PCV-1230	______
Comp.Cooling from RCP Motor Coolers	MOV-784/MOV-786	______
Comp. Cooling to RCP	MOV-797/MOV-769	______
PRT to Gas Analyzer	548/549	______
Makeup Water to PRT	552/519	______
Letdown Isolation	201/202	______
RCP Seal Return Isolation	MOV-222	______
RCS Sample Line	956E/956F	______
Accumulator Sample	956G/956H	______
Primary System Vent Header and H_2 Supply	1786/1787	______
RCDT to Gas Analyzer	1788/1789	______
RCDT Pumps to Holdup Tank	1702/1705	______
RCP Cooling Water Out	FCV-625/789	______
Excess Letdown Ht.Ex.Cooling Wtr.In	791/798	______
Excess Letdown Ht.Ex.Cooling Wtr.Out	796/793	______
Containment Sump Pump Discharge	1728/1723	______
Containment Air Sample In	1234/1235	______
Containment Air Sample Out	1236/1237	______
Stm. Gen. Blowdown	PCV-1214,1215,1216,1217	______
	PCV-1214A,1215A,1216A,1217A	______
Stm.Gen.Blowdown Sample	PCV-1223,1224,1225,1226	______
	PCV-1223A,1224A,1225A,1226A	______
Purge Supply Duct	FCV-1170/1171	______
Purge Exhaust Duct	FCV-1172/1173	______
Containment Pressure Relief	PCV-1190/1191/1192	______
Prz. Stm. Space Sample	956A/956B	______
Prz. Liquid Space Sample	956C/956D	______
O_2 Supply to Containment	IV-2A/IV-2B	______
H_2 Supply to H_2 Recombiner	IV-3A, 5A	______
	IV-3B, 5B	______
Instrument Air/P.A. Venting Supply Line	PCV-1228	______

4.3 All blind flanges listed below are installed.

1) H_2 Supply to H_2 Recombiner, between diaphragm and solenoid (4 lines). ______

2) Post Accident Venting Exhaust Line. ______

3) Containment Leak Test Instrument Line ______

4) Containment Leak Test Air Line ______

5) Residual Heat Removal Loop Out Between Valves 730 and 732 ______

6) Fuel Transfer Tube Blind Flange ______

7) Containment Spray Header Between Valves 869A and 867A and 869B and 867B ______

8) N_2 Supply to RCDT Between Valves 1610 and 1616 ______

9) Containment Air Sample In Between Valves 1234 and 1235 ______

10) Containment Air Sample Out Between Valves 1236 and 1237 ______

11) Air Ejector Discharge to Containment Between PCV-1229 and 1230 ______

12) Purge Supply Duct Between Valves 1170 and 1171 ______

13) Purge Exhaust Duct Between Valves 1172 and 1173 ______

14) Pressure Relief Line Between 1190 and 1191 and 1191 and 1192 ______

15) Supply to Post Accident Sampling Lines Between 1890F and 1890C ______

16) Post Accident Return Sample Lines Between 1890G and 1890H ______

17) O_2 to Containment - Downstream of IV-2A and 1882A ______

4.4 Check Valves listed below are operable.

518	N_2 Supply	______
741	RHR Pumps to RHR Heat Exchanger	______
867A	To Containment Spray Header	______
867B	To Containment Spray Header	______
1616	N_2 Supply to RCDT	______
IA-39	Instrument Air to Containment	______

4.5 The Equipment Door is properly closed. ______ 1

4.6 At least one door in each personnel air lock is closed. (Verify by lights in Control Room.) ______

4.7 Containment Air Leakage meets the requirements of the integrated leak rate test. ______ 1

Containment Integrity Established

Date ____________________

Time ____________________

WF ____________________

INDIAN POINT STATION

UNIT NO. 3

PEP-ES-1A REV. 1

LOSS OF COOLANT TO CONTAINMENT

2.5 With regard to containment isolation, the following must be initiated:

2.5.A When the charging pumps are no longer required, have the Nuclear Plant NPO isolate the charging line by closing valves 205, 226 and 227 and admit isolation valve seal water by opening 1402.

2.5.B After Reactor Coolant Pump No. 31 is stopped, have the Nuclear Plant NPO close its' injection water supply isolation valves (241A and 250A) and admit isolation valve seal water by opening 1466.

2.5.C After Reactor Coolant Pump No. 32 is stopped, have the Nuclear Plant NPO close its' injection water supply isolation valves (241B and 250B) and admit isolation valve seal water by opening 1467.

2.5.D After Reactor Coolant Pump No. 33 is stopped, have the Nuclear Plant NPO close its' injection water supply isolation valves (241C and 250C) and admit isolation valve seal water by opening 1468.

2.5.E After Reactor Coolant Pump No. 34 is stopped, have the Nuclear Plant NPO close its' injection water supply isolation valves (241D and 250D) and admit isolation valve seal water by opening 1469.

2.5.F After all Reactor Coolant Pumps are stopped,

1) Isolate the No. 1 seal water return line by closing seal return valves 261A, 261B, 261C, 261D and MOV-222. Manually initiate isolation valve seal water to MOV-222 by opening valve 1418.

2) Isolate the Reactor Coolant Pumps cooling water supply and return lines (MOV-769, MOV-797, MOV-786, MOV-784, MOV-789 and MOV-625). Manually initiate isolation valve seal water to MOV-797, 784 and 625 by opening valves 1421, 1416 and 1417 respectively.

NOTE: These valves are operated from the Containment Isolation panel and may already be closed as the result of a high-high containment pressure condition which generates a containment isolation Phase B signal. Automatic seal water injection to these valves will have been initiated by the earlier Containment "Phase A" Isolation signal.

2.5.G If the safety injection test line is in service, have the Nuclear Plant NPO close its' manual containment isolation valves (859A and 859C).

2.5.H Following the shutting down of both Containment Spray Pumps have the Nuclear Plant NPO close the containment spray lines manual containment isolation valves (869A and 869B) and introduce isolation valve seal water by opening 1405 and 1463.

2.5.I If on low head recirculation:

1) Have the Nuclear Plant NPO close manually operated High Head Safety Injection Pump 31 Discharge valve (850A). Close High Head Safety Injection Pump Discharge Valve MOV-851A and have the Nuclear Plant NPO introduce isolation valve seal water to both valves by opening 1464 and 1465.

2) Assure that the valves from the low head to the high head path, MOV-888A and MOV-888B are closed and have the Nuclear Plant NPO introduce isolation valve N_2 seal gas by opening 1447 and 1448.

3) Close motor operated valves 1835A and 1835B on the outlet of the boron injection tank. Introduce isolation valve N_2 seal gas to each valve by opening 1479 and 1403.

2.5.J If the Residual Heat Removal pumps are not in service, check closed their motor operated discharge valve and mini-flow valves (MOV-743, 744 and 1870) and have the Nuclear Plant NPO assure that the RHR sample valve (958) in the valve room is closed. Have the Nuclear Plant NPO introduce isolation valve N_2 seal gas to each by opening 1401, 1446, 1449 and 1450.

2.5.K Have the Nuclear Plant NPO assure that the Recirculation Pump sample valves (990B and 990C) in the valve room are closed and introduce isolation valve N_2 seal gas by opening 1420.

2.5.L Have the Nuclear Plant NPO close the Nitrogen supply lines to the Pressurizer Relief Tank and Reactor Coolant Drain Tank (550 and 1610) in the valve room.

2.6 Sample the containment atmosphere for hydrogen concentration on a daily basis and plot a curve of hydrogen generation vs. time. Use this information to determine the need for the hydrogen recombiner (SOP-CB-7) or the post accident containment venting system (SOP-CB-6).

Supplemental Response to Item H.5

To supplement our response of November 12, 1975 to Item H.5., actual measurements of sample line temperatures were taken at Indian Point Unit No. 2 with various flow variations between lines during the period of November 14 through November 17. All measurements taken verified that the design limit ΔT (ΔT defined as the difference between the hottest line and the average of all lines) was met.

Our previous response presented only the allowable ΔT (100°F) for design conditions of 650°F. For lower actual temperatures, the allowable ΔT is correspondingly higher due to improved material properties. In all observed measurements the resultant maximum sample line stress was within the calculated allowable code stress. Further, a fatigue analysis performed on the maximum observed variation in temperature due to flow variations over a several hour period demonstrates that in excess of 10^6 such cycles are acceptable before code limits are reached.

For the above reasons, we conclude that planning sample line penetration modifications to be completed during the first convenient outage following power operation but no later than the first refueling outage will have no adverse effect on the safety of the facility.