ATTACHMENT
(REFERENCE ITEM 2A)

EVALUATION OF CHANNEL HEAD FLAW GROWTH BY CORROSION FATIGUE

CONSOLIDATED EDISON INDIAN POINT UNIT NO. 3
STEAM GENERATOR
CHANNEL HEADS

BY

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EVALUATION OF CHANNEL HEAD FLAW GROWTH BY CORROSION FATIGUE

1.0 Summary

This report presents an evaluation of the possible growth of the imperfections identified in the Consolidated Edison Indian Point Unit No. 3 steam generator channel head cladding. The evaluation is based on the "Rules for Inservice Inspection of Nuclear Power Plant Components" presented in Section XI of the ASME Boiler and Pressure Vessel Code.

The flaw growth assessment is made based on the stresses in that region of the channel head which is the highest stressed area known to contain imperfections.

Critical flaw sizes are determined for assumed flaws of aspect ratios ranging from zero to 0.2. Further, it is assumed that a semi-elliptical flaw extending into the base metal is present in the "highest stress" region, and the initial sizes of this hypothetical flaw which meet the acceptability criteria of Section XI paragraph IWB-3600 are determined.

It is conservatively concluded that an initial flaw with an aspect ratio equal to 0.0 (i.e. of infinite length) could be as deep as 0.116" into the base metal, and would not exceed one-tenth the critical flaw size at the end of the service life of the steam generator. Furthermore, initial flaws with aspect ratios greater than zero, as show in Figure 1.1, could be deeper then 0.116" into the

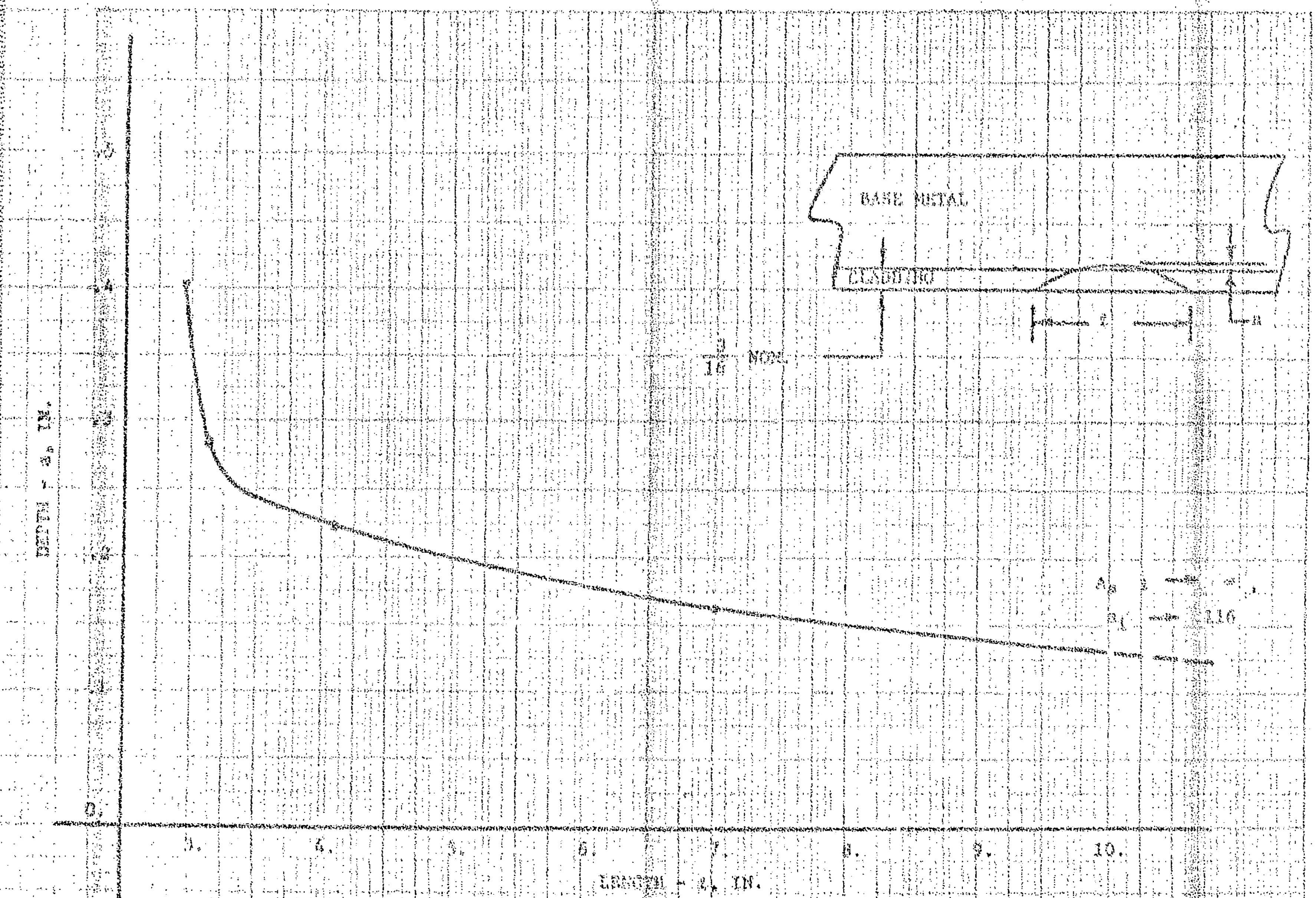


Figure 1.1. Typical Flaw Depth vs. Length

base metal, and similarly would not exceed one-tenth the critical flaw size at the end of the service life of the steam generator.

It is known that none of the imperfections in the cladding of the channel heads in the Indian Point Unit No. 3 steam generators have an aspect ratio equal to zero, and none have been identified as extending any depth into the base metal. Therefore, it is concluded that the integrity of the channel heads will not be violated during the 40 year service life of the steam generators.

2.0 Steam Generator Duty Cycle

The specified cyclic duty of the Indian Point Unit No. 3 "44" Series Steam Generator is tabulated in Table 2.1 together with that of a typical "51" Series Steam Generator. The analysis following is performed using the number of cycles specified for the "44" Series Steam Generator and the stresses calculated for the "51" Series. Series "51" steam generator stresses are used instead of the actual Series "44" steam generator stresses because the improved techniques for computations has made more detailed data available for the "51" Series than are available for the "44" Series.

The "44" series steam generator operates at a somewhat higher temperature than the "51" series steam generator, specifically 612.6 F versus 605 F and 554.8 F versus

TABLE 2.1
(Sheet 1 of 2)

Load States

Condition	51 Series Event	51 Series Number of Occurrences	44 Series Event	44 Series Number of Occurrences
Normal	Plant Heat Up	200	Plant Heat Up	200
	Plant Cool Down	200	Plant Cool Down	200
	Plant Loading	18,300	Plant Loading	14,500
	Plant Unloading	18,300	Plant Unloading	14,500
	Small Step Load Increase	2,000	Small Step Load Increase	2,000
	Small Step Load Decrease	2,000	Small Step Load Decrease	2,000
	Step Reduction from 100% to 50%	200	Step Reduction from 100% to 50%	200
	Hot Standby Operation	18,300	Hot Standby Operation	25,000*
	Turbine Roll Test	10	Not Specified	
Upset	Reactor Trip from 100%	400	Reactor Trip from 100%	400
	Loss of Load	80	Loss of Load	80
	Loss of Flow	8	Loss of Flow	80
	Loss of Power	40	Loss of Power	10
	None Specified		Loss of Secondary Pressure	6
	OBE	5 of 10	"g" loading & normal load	steady state

*25,000 cycles represents manual S.G. water level control. No significant cyclic effect @ T/S locale.

TABLE 2.1 (CONT.)
(Sheet 2 of 2)

Load States Cont.

Condition	51 Series Event	Number of Occurrences	44 Series Event	Number of Occurrences
Tests	Primary Hydrostatic	5	Primary Hydrostatic	1
Tests	Secondary Hydrostatic (0 psig primary)	5	Secondary Hydrostatic	15
Tests	None Specified		Primary Pressure Tests	5
Tests	None Specified		Primary Leak Test (2250/0)	5
Tests	None Specified		Secondary Leak Test (0/840)	5
Emer.	Design Basis Earthquake (DBE)	3 of 10	None Specified	
Faulted	Reactor Coolant Pipe Break (LOCA)	1	None Specified	
Faulted	LOCA + DBE	1	(g) Loading	
Faulted	Stream Line Break + DBE	1	(g) Loading + 2485/0	Steady State

Reference: Westinghouse Equipment Specification 676219, Rev. 1 dated 12/26/67,

Project: Consolidated Edison Co.

542 F for primary coolant inlet and outlet, respectively. These temperature differences are not considered significant.

In general, the transient responses are similar, with the "51" Series response usually being somewhat more severe. It is concluded that an assessment of the stress history of a steam generator which has been subject to the "51" Series thermal history will conservatively represent the "44" Series.

The dimensions relevant to a structural analysis of the high stressed region at the channel head (cylindrical section above biological shield, adjacent to tubesheet) are shown in Figure 2.1. The primary bearing sections of interest are:

Channel Head wall adjacent to tubesheet:

	"44"	"51"
Thickness	5.25"	5.16"
Mean Radius	62"	63.39"

Secondary sheel adjacent to tubesheet

T.S. Thickness	22"	21.03"
Thickness	3.25"	3.25"
Mean Radius	63"	66.345'

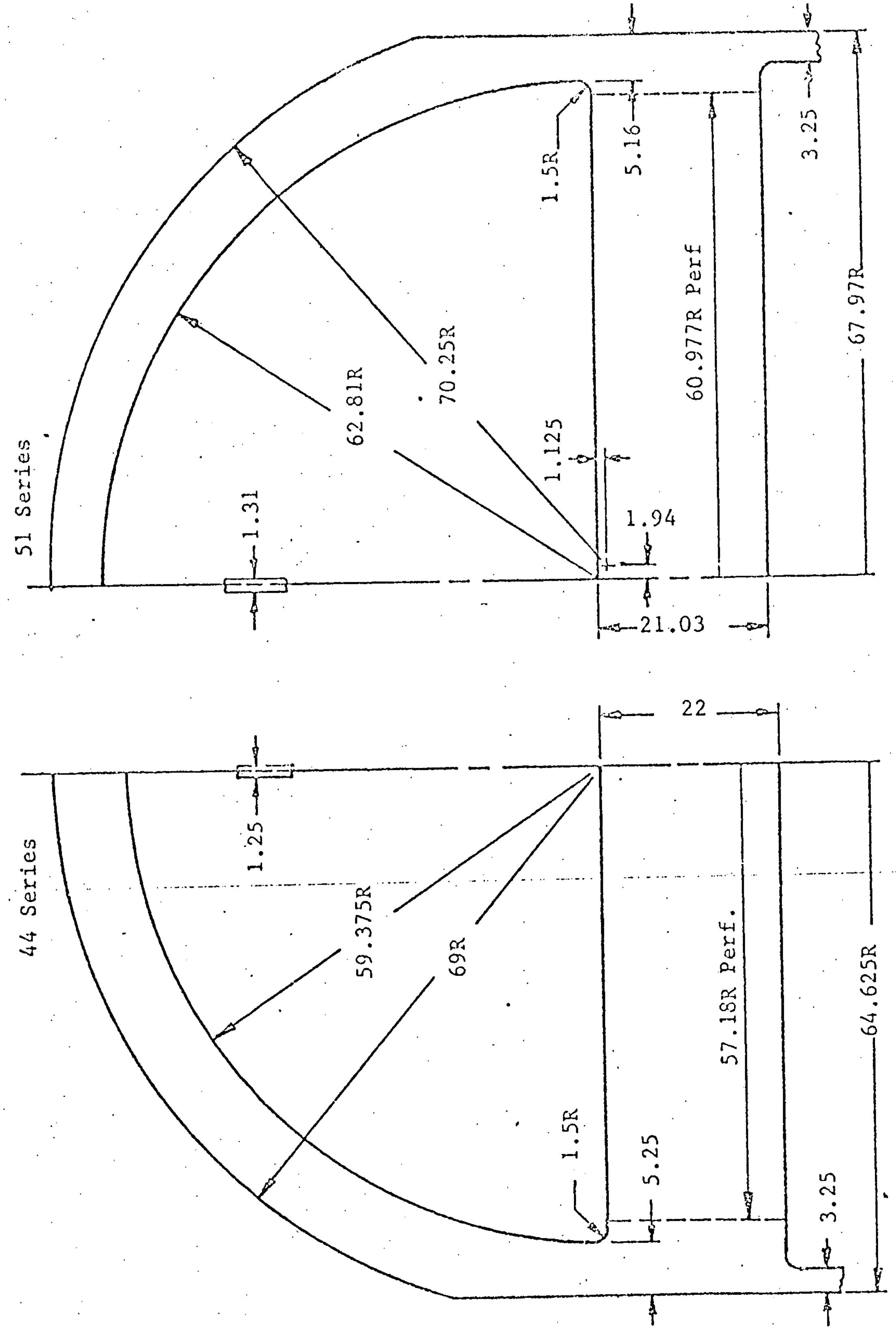


Fig 2.1

Comparative Sketches of "44" and "51" series steam generator channel heads. (No cladding shown)

It is apparent from the above table and operating pressures that the "51" Series has greater primary loading than the "44" Series and, therefore, use of "51" Series stresses is a conservative approach. The secondary stresses, which are section thickness dependent, are very similar.

3.0 Critical Flaw Size

The critical flaw size in the channel head was determined at the tube sheet-to-head weld region. This region was selected because it displayed higher stresses than any of the other regions where indications were found.

The computations of the critical flaw size is based on the maximum state of stress in the channel head for the area of concern. This occurs during the primary hydrotest. Although the equipment specifications provide for only one initial primary hydrotest (which has already been performed), the analysis assumes that an additional primary hydrotest at 3125 psia would be conducted. The lowest temperature at which any subsequent proof testing would be conducted is above 284 F (Reference: Heat-up Curves, Indian Point Unit No. 3 Proposed Technical Specifications).

Figure 3.1 is a plot of flaw depths of aspect ratios (a/l) ranging from 0.0 to 0.2 and the corresponding stress intensity factors (K_I). The critical flaw size is that flaw

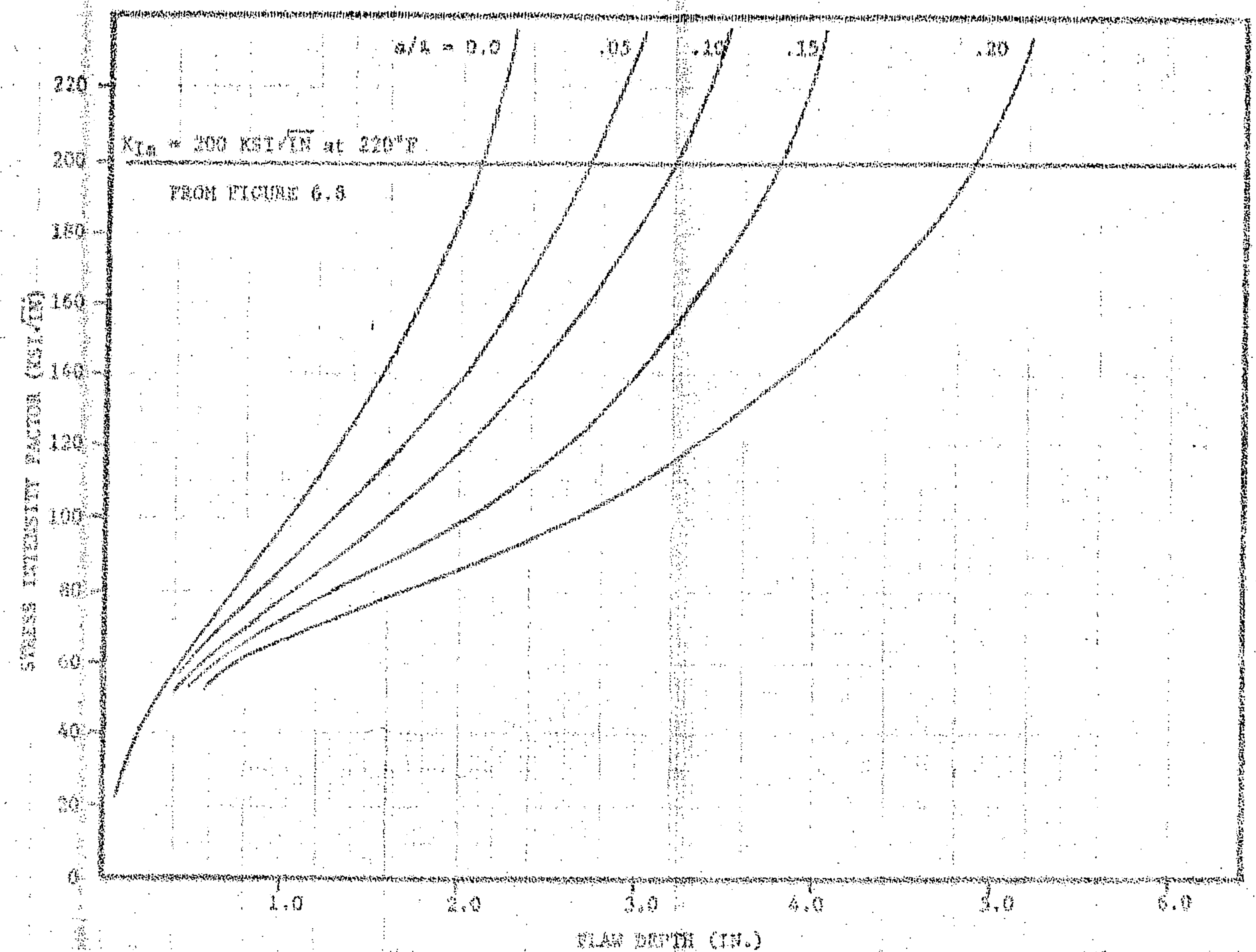


Figure 3.1. Critical Flaw Size

size at which the calculated stress intensity factor (K_I) equals the material fracture toughness (K_{Ia}). Based on this, Figure 3.2 presents the relationship between aspect ratios and critical flaw size.

4.0 Analysis of Flaw Growth

The transients listed in Table 2.1 were grouped as shown in Table 4.1 on the basis of pressure loads and stress levels. Within each grouping, each event was assumed to occur at the highest stress for the entire group. Specifically, all seventeen events of Group I were assumed to occur at the initial primary hydrotest pressure. Loss of load was taken as the governing transient for Group IV, reactor trip for Group V, and large step load decrease for Group VI.

After grouping the transients and determining the number of occurences for each group, the subcritical flaw growth was computed assuming an initial flaw of at least the nominal cladding thickness, and iterating until the maximum allowable initial flaw size was determined. The maximum allowable initial flaw size is that flaw size which will grow to one-tenth of the associated critical flaw size by the end of the 40-year design life.

It is necessary to assume that a flaw exists in order to compute a stress intensity factor range (ΔK_I) for use in the flaw growth equation. It is emphasized that no flaw

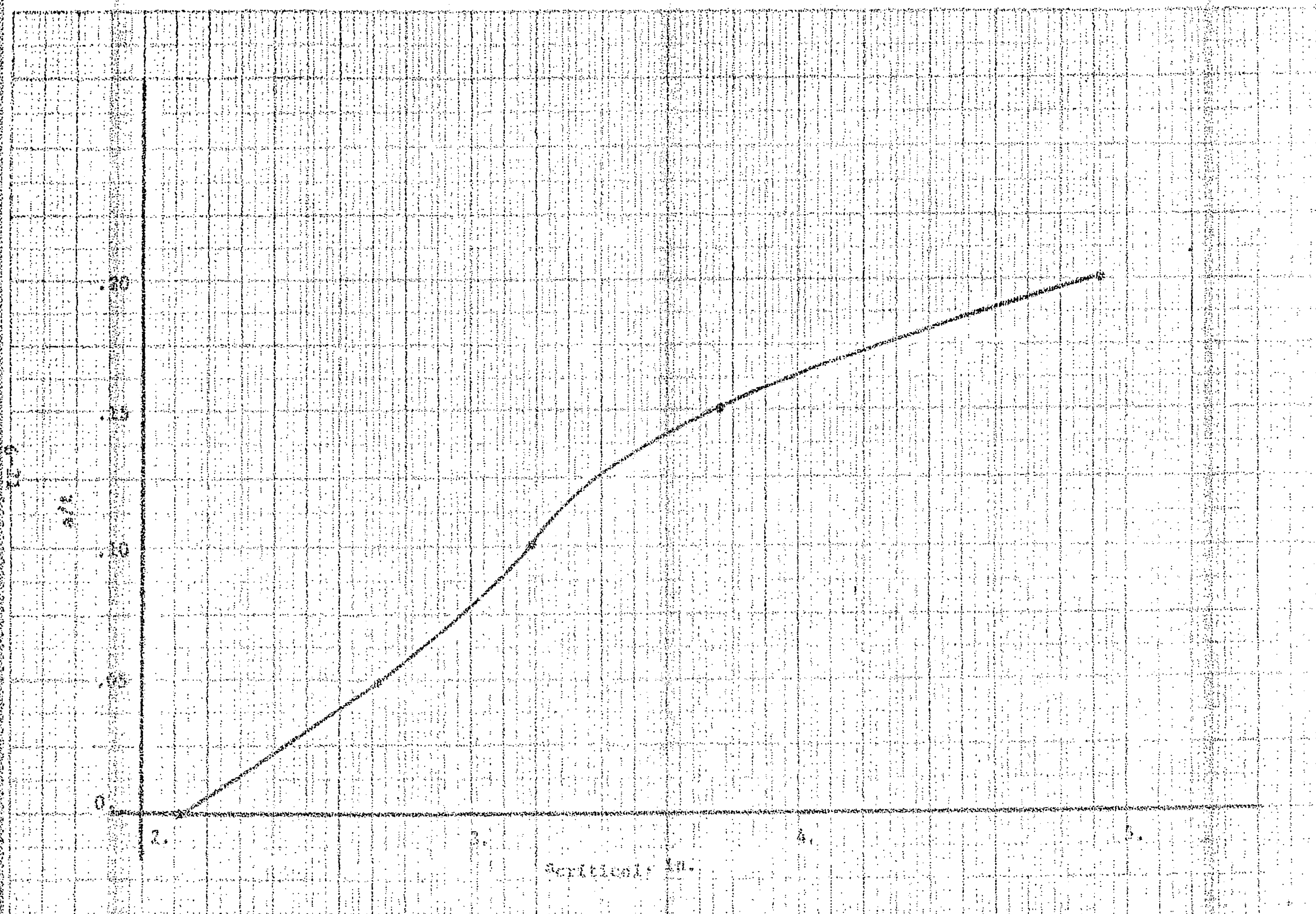


Figure 3.2. Aspect Ratio vs. Critical Flaw Size

TABLE 4.1

TRANSIENT GROUPS

Transient Group	E-Spec. Transient	Occurrences
I	Initial Primary Hydro Test	1
	Subsequent Primary Hydro Test	5
	Primary Leak Tests	5
	Loss of Secondary Power	6
		17
II	Heat up and Cool down	200
III	Plant Loading and Unloading	13,930
IV	Loss of Load	80
	Loss of Flow	80
		160
V	Reactor Trip	400
	Loss of Power	10
		410
VI	Large Step Load Decrease	200
	10% Step Load Increase	2,000
	10% Step Load Decrease	2,000
		4,200

was detected in the channel head base metal and that the ssumption of a flaw is an analytical device used to evaluate possible flaw growth.

The equation used for flaw growth in SA216 Grade WCC/ SA533 Grade A-Class 1 heat affected zone material in an air environment is $da/dN = 4.77 \times 10^{-11} \Delta K_I^{3.55}$ (Reference: W. A Logsdon and W. H. Pryle, "Fracture Toughness and Fatigue Crack Growth Rate Material Properties of A216 (WCC Grade)/A533 Grade A Class 1 and A533 Grade A Class 11 Submerged Are Weldments", Westinghouse Research Report 73-1E7-FCAST-R, December 31, 1973). This equation is corrected for a PWR environment by using the PWR/air environment correlation given for da/dN in Section XI, Article A-4000, Figure A-4300-1, in which the flaw growth rate in a PWR environment is taken as 14.2 times the growth rate in air. Multiplying the da/dN equation above by 14.2 gives $da/dN = 6.77 \times 10^{-10} \Delta K_I^{3.55}$ for SA216 Grade WCC/ SA533 HAZ in a PWR environment. This value for da/dN, as compared to the flaw growth equation for SA508 steels presented in Article A-5000 of Section XI s conservatively high in the ΔK_I ranges considered.

The end of life flaw size was calculated to be less than one-tenth the critical flaw size for each a/l ratio, with both the final flaw size and the critical flaw size measured from the based metal/cladding interface. Details of the calculations for an assumed flaw with an aspect ratio of 0.0 are given in Table 4.2

5.0 Evaluation of Data

Based on a conservatively high flaw growth rate, it has been calculated that an initial flaw extending into the base metal and as large as indicated in Figure 1.1 would not exceed one tenth the critical flaw size of the channel heads at the end of the 40 yr. service life of the steam generators at Indian Point Unit No. 3.

These values are conservative because:

1. The assumption of the presence of a flaw was necessary for analytical projections. Actually no such flaw is known to be present. In the course of the attempt to remove some imperfections by grinding away the cladding, it was established that some imperfections extended to, but not into the base metal. This was further corroborated in the evaluation of the boat samples taken from the steam generator channel heads. Furthermore, examination and evaluation of conditions at the basemetal/cladding interface indicate low probability of propogation of existing imperfections into the base metal during the service life of the steam generators. (This is discussed in detail in Subsection 6.1 of the "Technical Report on Steam Generator Channel Head Cladding, Indian Point Unit No. 3" dated September 19, 1975.)

TABLE 4.2

SUBCRITICAL FLAW CROWTH
$a/l = 0.0$

Transient Group	Governing Transient	σ_m KSI	σ_b KSI	$\frac{\sigma_m+\sigma_b}{\sigma_y}$	a*	a/t	M_m	M_b	Q	K_I**	K_I	$\frac{da}{dN}$	N	da
II	Heat-up	11.7	0.8	0.6	0.3	.057	1.15	1.06	.900	14.58	14.58	9.16×10^{-6}	200	.00183
III	Plant Load	11.5	9.7	0.6	0.3	.057	1.13	1.06	.900	23.97	9.39	1.02×10^{-6}	13930	.02676
IV	Loss of Load	11.0	25.0	1.0	0.327	.062	1.15	1.05	.793	44.38	29.80	1.16×10^{-4}	160	.01834
V	Reactor Trip	8.6	23.4	1.0	0.346	.066	1.15	1.05	.793	40.42	25.84	6.99×10^{-5}	410	.02866
I	Primary Hydro	16.0	26.0	1.0	0.375	.071	1.16	1.05	.793	35.03	35.83	1.08×10^{-3}	17	.01830
VI	Large Step	12.2	10.3	0.6	0.393	.075	1.16	1.05	.900	29.24	5.27	2.47×10^{-7}	4200	.00104
														.09512

$K_I = \sigma_m M_m \sqrt{\pi}\sqrt{a/Q} + \sigma_b M_b \sqrt{\pi}\sqrt{a/Q}$

* Initial flaw depth determined by iteration. Depth of flaw includes 0.188 thickness of cladding

** Article A-3000, Section XI

2. Series "51" steam generator stresses are used instead of the actual Series "44" steam generator stresses. Furthermore, transients were grouped, and within each group each transient was assumed to occur at the highest stress for the group. Also, the sequence of transients was taken so that the cumulative effect on flaw growth was maximized.

6.0 Conclusions

In the course of the investigation of the imperfections in the cladding of the Indian Point Unit No. 3 steam generator channel heads, it was determined that the longest imperfection was of the order of several inches, and that none of the imperfections extended into the base metal.

The analysis presented herein conservatively establishes sizes of flaws that could be present in the highest stress region of the channel head and still would not exceed one-tenth the critical flaw size at the end of the 40 year service life of the steam generators.

Therefore, it is concluded that growth of any of the imperfections now present in the Indian Point Unit No. 3 steam generator channel heads will not affect the integrity of the channel head during the 40 year service life.

ENCLOSURE 2

Item 2. Steam Generator Cladding

(b) Surface Examination Techniques

In the "Technical Report on Steam Generator Channel Head Cladding, Indian Point Unit No. 3" dated September 19, 1975, it was proposed, as part of the surveillance program (Ref: Section 10) to increase the ASME B&PV Code Section XI requirement for visual examination of 36 square inches to a general 100% visual examination of the interior water box surface per inspection interval. In practice, such visual examination could be accomplished remotely by use of a TV camera, thereby minimizing exposure of personnel to radiation, and making it possible to "zoom-in" on questionable areas for closer examination.

It is now proposed to accomplish the general 100% visual examination at each of the first three refueling shutdowns. In the event that signs of deterioration of the cladding are noted, questionable areas will be liquid penetrant examined, and, where surfaces are suitable for replication a material such as RTV-11 silicone rubber will be used to obtain replications for further study and measurement.

Photographs will be taken of any indications developed by the liquid penetrant examination and will be compared to photographs of cladding indications now available to Con Edison. The photographs will be used to help evaluate the extent of deterioration, if any, and to determine the need for further action.

It was concluded in the Con Edison September 19, 1975 report that additional imperfections in the cladding may develop, and that existing imperfections may propogate within the cladding. However, it was anticipated that the rate of formation and propogation will be insignificant. It was also concluded that conditions at the base metal/cladding interface indicated low probability of propagation of imperfections into the base metal during the service life of the steam generators. (Ref: Subsection 5.3.1)

ENCLOSURE 2

Item 2. Steam Generator Cladding

(c) Ultrasonic Examination Development Program

Con Edison plans to continue the development and system verification of the ultrasonic method of surveillance examination. The objective of the continuing study is to refine the definition of the ultrasonic system's sensitivity and fidelity and to reduce the size of the minimum detectable flaw. The results of this development program will be incorporated into procedures to be used at the first three refueling shutdowns.

There is available to Con Edison a representative section from a channel head casting in which a series of notches have been machined. Additional sections are available from Westinghouse Tampa. Some of the parameters which will be explored are:

ultrasonic examination frequency

angle between crystal and work

type and size of crystal.

Work will also be done on the preparation of test pieces with various simulated defect sizes and angles. Defect generation processes will also be tested to more closely simulate the imperfections that presently exist in the cladding.

ENCLOSURE 2

Item 2 - Steam Generator Cladding

(d) Technical Specification for Inservice Inspection

Con Edison requests that Section 4.2 of the Technical Specifications for Indian Point Unit No. 3, when issued, contain the following information:

Item 3.7 (Category I-2) Steam Generator Head - Weld Cladding

3.7.1 General

The exterior surfaces of the head are accessible by removal of covering insulation. The surfaces to be examined by ultrasonic testing during surveillance test operation have been prepared and specific location of test sites are noted in Table 4.2-1.

The interior areas of the primary sides of the steam generator are accessible through manways provided on the inlet and outlet sides of the head.

All the examinations scheduled are listed in Table 4.2-1.

3.7.2 Visual Examination

Visual examination of the interior surfaces of the head will be performed during each of the first three refueling shutdowns.

Evaluation of the results of the prior examinations will be utilized to establish the scope of subsequent visual examinations.

3.7.3 Liquid Penetrant Test

Specific sections of the internal weld surfaces of the head, which after visual examination, require more definitive investigation shall be evaluated by liquid penetrant testing. Additional non-destructive testing including replication using RTV-11 silicone rubber type material will be employed where liquid penetrant results indicate that they are required.

3.7.4 Ultrasonic Test

Monitoring of the existing welded cladding conditions shall be performed by ultrasonic testing from the exterior surface. Eight sectors on three steam generator heads at the locations listed in Table 4.2-1 shall be surveyed during each of the first three refueling outages using the ultrasonic test process previously qualified. The results will be compared to base line data generated prior to reactor start-up. Evaluation of the results following the third refueling outage will determine whether a test program should be continued for subsequent shutdowns.

Table 4.2-1

Item No.	Examination Category	Components and Parts to be Examined	Method	Extent of Examination	Remarks
3.7	I-2	Steam Generator Head welded cladding surfaces - primary sides.	Visual	100%	To be performed during first three refueling shutdowns.
		"	Liquid Penetrant	Sample	As warranted by results of visual examination.
		"	Alternate NDT or repli- cation by RV-11		As warranted by results of visual and liquid penetrant examination.
		Steam Generator Head welded cladding surfaces - primary sides and eight sectors on Steam Generators #31, #32 and #34 per Figure 1 & 2.	Ultra- sonic Test		Ultrasonic testing will be performed during first three refueling shutdowns.

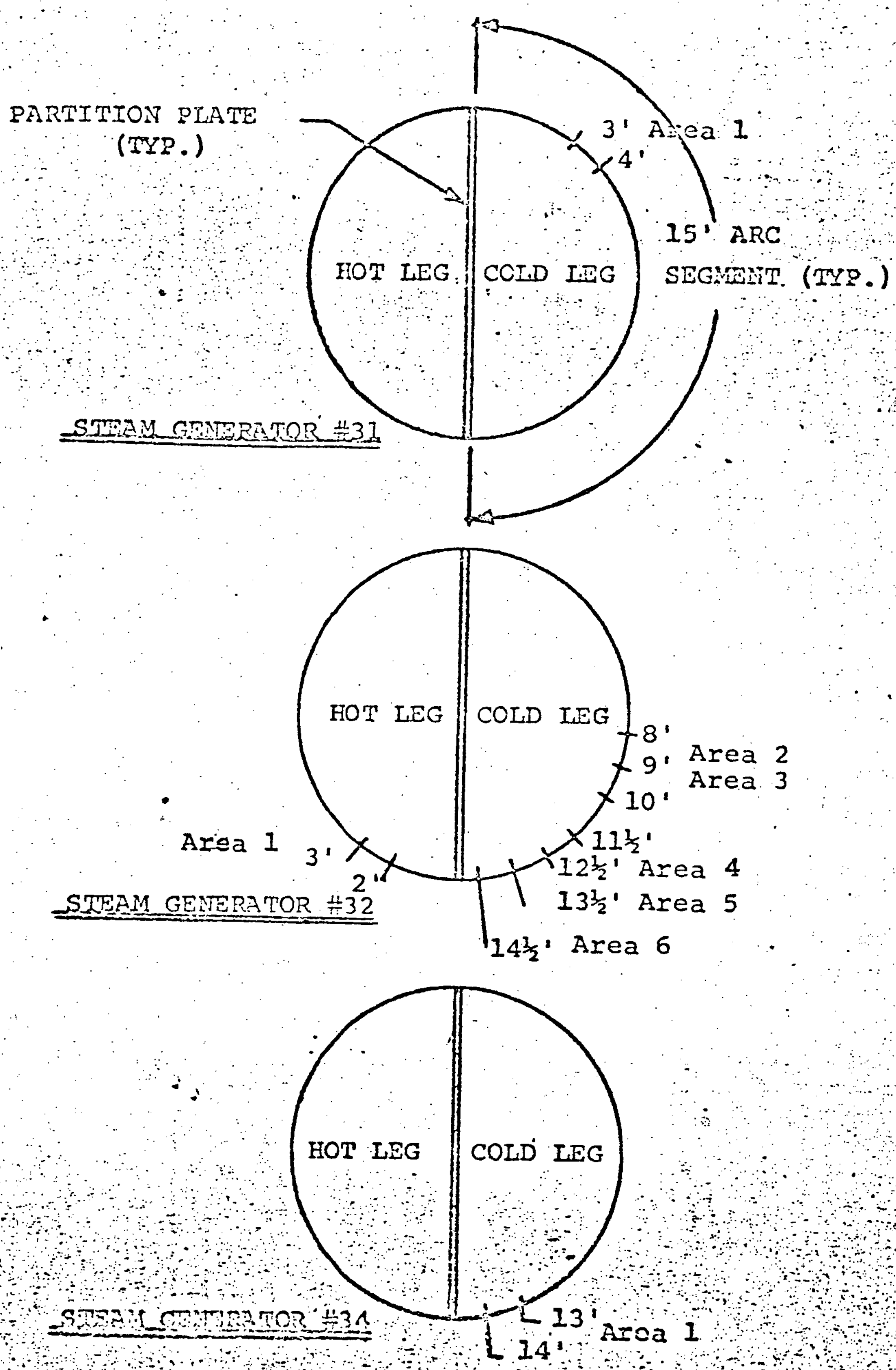

PARTITION PLATE
(TYP.)
3' Area 1
4'
15' ARC
SEGMENT (TYP.)
HOT LEG
COLD LEG
STEAM GENERATOR #31
HOT LEG
COLD LEG
8'
9'
Area 2
Area 3
10'
11½'
12½' Area 4
13½' Area 5
14½' Area 6
Area 1
3'
2'
STEAM GENERATOR #32
HOT LEG
COLD LEG
13'
14'
Area 1
STEAM GENERATOR #34

FIGURE 1

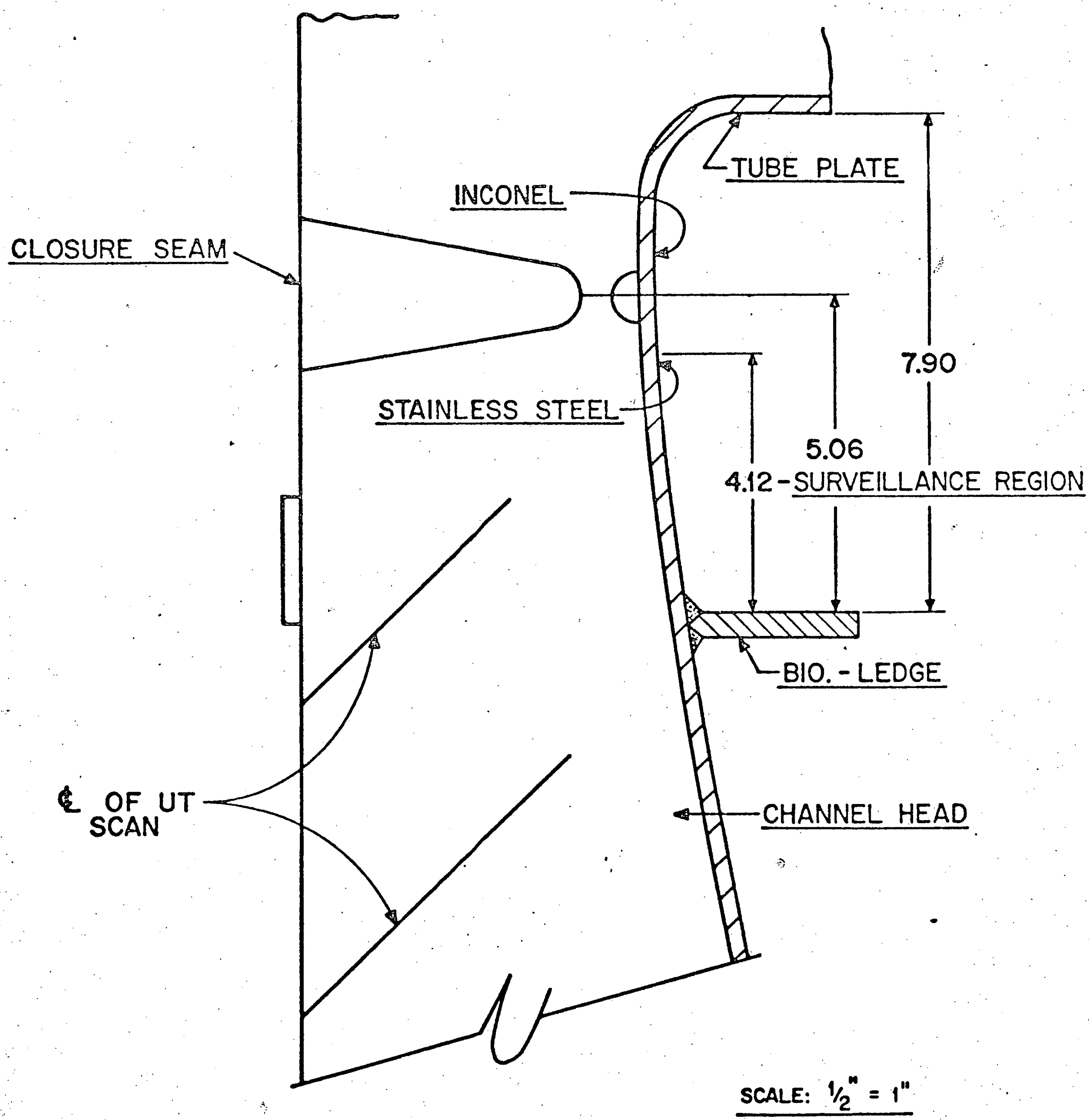


FIG. 2 SURVEILLANCE REGION

Item 3. EMERGENCY CORE COOLING

(a) Sensitivity Studies

No response required.

(b) Boron Precipitation

Our procedure for initiating hot leg recirculation is presented in the response to Item 3c(ii). The procedure provides instructions for initiating hot leg recirculation approximately 21 hours after the postulated accident.

(c) Submerged Valves

(i) After a postulated loss-of-coolant accident, the water level rise inside containment has been calculated to reach El. 50'1-1/2". This flooding level is based on the contents of the Refueling Water Storage Tank, the Spray Additive Tank, the four accumulators and approximately 1/3 of the reactor coolant system being emptied into containment.

Those remote operated valves which may become submerged following a postulated loss-of-coolant accident and the consequences of their being flooded considering both short-term and long-term ECCS functions, containment isolation and other safety functions are presented in the attached table.

During the injection phase following a postulated loss-of-coolant accident, all of the cold leg injection lines must be open (Valves 856A, C, D, E, F, H, J and K) and the hot leg injection lines closed (Valves 856B and G) in order to assure that the flow required by the ECCS analysis is delivered to the core assuming a single failure. The location of these valves in the fluid system is shown on FSAR Figure 6.2-1. In order to assure this mode of operation, the cold leg valves will be de-energized at their motor control center in the open position. The hot leg valves will be similarly de-energized in the closed position. Separate position indication for these valves will be provided in the control room to assure that valve status is in the proper safeguards position for the depowered condition.

At approximately 21 hours after the postulated loss-of-coolant accident, hot leg recirculation will be initiated in order to assure that boric acid concentration in the core does not reach unacceptable levels. To establish hot leg recirculation in each high-head safety injection train, two of the four cold leg injection paths must close and the hot leg injection path open. To effect this realignment, the motor operator on Valves 856B, C, E, G, H and J will be relocated remote from the valve to a position above Elevation 51'0". The valves will be operated via a positive mechanical drive arrangement which incorporates flexibility by use of splined couplings and universal joints. A conceptual sketch of the valve operator relocation is shown in Figure 1. The frame supporting the valve operators will be designed as a seismic Class 1 structure.

For each of these valves, a specific design based on this concept will be developed. The components used in the design are simple mechanical devices used extensively in many similar applications. As such, their reliability is expected to be very high. The materials used in these components have been or will be evaluated for service in containment following the postulated LOCA. The suitability of these components will be verified by a three-day submergence test under LOCA pressure, temperature and chemistry conditions.

The valve control logics will be modified to assure that each hot leg valve cannot be opened until the two relocated operator cold leg valves have been closed in that safety injection train. This will prevent the safety injection pumps from exceeding their maximum flow limit. The control logic on the two cold leg valves in each safety injection train whose operator has not been relocated will be de-activated.

The modifications proposed do not entail any changes in the piping system, and therefore the results of the pre-operational high-head flow tests or other functional tests are not affected.

The system with the proposed changes can withstand any of the following single failures and still deliver the flow to the core as required by the ECCS analysis:

a. failure of a diesel generator to start

b. failure of a high-head pump to start

c. passive failure in pump discharge piping during the recirculation phase

d. failure of a valve to operate

e. passive failure in pump suction piping during the recirculation phase

In summary, the changes being proposed are as follows:

(a) Relocate the motor operator on Valves 856B, C, E, G, H and J.

(b) De-energize the valves listed below at their motor control center in the position indicated.

856A, C, D, E, F, H, J and K (Open)
856B, and G (Closed)

(c) Modify the valve control logic on the relocated operators to prevent the hot leg valves from being opened before the two cold leg valves in that SI train are closed.

(d) Deactivate the control logic on the non-relocated cold leg valves.

(e) Provide separate position indication in the control room for both the relocated and non-relocated valve operators for valves listed in Item (b) above.

(ii) Attached is an extract from System Operating Procedure (PEP-ES-1A, Rev. 1), Loss-of-Coolant to Containment", which identifies the steps required to initiate hot leg recirculation.

Attached is a copy of the affected electrical schematics (500B971 sheets 112, 117 and 155) marked up to reflect the proposed design changes.

INDIAN POINT UNIT NO. 3

SUBMERGED VALVE OPERATORS WITHIN CONTAINMENT (Page 1 of 5)

[BASED ON POST-LOCA MAXIMUM FLOOD LEVEL AT ELEVATION 50' 1-1/2"]

*Short Term-Injection Phase of LOCA **Recirculation Phase of LOCA

Valve	Function	Valve Position During Plant Operation	Assumed Consequences to ECCS Performance Due to Flooding of Valve Operator: Short Term*	Assumed Consequences to ECCS Performance Due to Flooding of Valve Operator: Long-Term**	Containment Isolation or Other Safety Related Functions	Design Change Proposed
856A	Supply Hi-Head Safety Injection Flow to R.C.S.) Cold leg during Post-LOCA Period (Loop #1)	Open	None	Valve Operator Malfunctions (eg, early spurious closure or fails to close) Transfer from cold leg to hot leg recirculation to prevent hi-levels of boric acid concentrations in core region (per Con Ed letter of 10/10/75) not completely accomplished.	None	De-activate control logic with valve in open position.
856D	Ditto (Loop #2)	Open	None	Ditto	None	Ditto
856 F	Ditto (Loop #3)	Open	None	Ditto	None	Ditto
856 K	Ditto (Loop #4)	Open	None	Ditto	None	Ditto
856 C	Ditto (Loop #4)	Open	None	Ditto	None	Relocate Operator

Valve	Function	Valve Position During Plant Operation	Assumed Consequences to ECCS Performance Due to Flooding of Valve Operator		Containment Isolation or Other Safety	Design Change Proposed
			Short-term	Long-term		
856 E	Ditto (Loop #1)	Open	None	Ditto	None	Relocate Operator
856 H	Ditto (Loop #3)	Open	None	Ditto	None	Ditto
856 J	Ditto (Loop #2)	Open	None	Ditto	None	Ditto
856B	Supplies Hi-Head Safety Injection flow to R.C.S. hot leg during post-LOCA period (Loop #3)	Closed and de-powered per requirements of Tech. Specs.	None	Valve Operator Malfunction (eg, fails to open) — Transfer from cold leg to hot leg recirculation to prevent hi-levels of boric acid in core region [per Con Edison letter of 10/10/75] not completely accomplished.	None	Ditto
856G	Ditto (Loop #1)	Ditto	None	Ditto	None	Ditto
123	Excess Let-Down Control - CVCS	Open	None - No requirement for Operation	None - No requirement for Operation	None	No

Valve	Function	Valve Position During Plant Operation	Assumed Consequences to ECCS Performance Due to Flooding of Valve Operator		Containment Isolation or Other Safety Related Functions	Design Change Proposed
			Short-term	Long-term		
200C	Letdown Orifice Isolation - CVCS	Open (Intermittent)	None - No requirement for Operation	None - No requirement for Operation	None	No
213	Auxiliary Spray Valve - CVCS	Ditto	Ditto	Ditto	None	No
890A	Accumulator #31 Fill Valve	Closed	Ditto	Ditto	None	No
890B	Accumulator #32 Fill Valve	Ditto	Ditto	Ditto	None	No
890D	Accumulator #34 Fill Valve	Ditto	Ditto	Ditto	None	No
891B	Accumulator #32 N_2 Fill Valve	Ditto	Ditto	Ditto	None	No
891D	Accumulator #34 N_2 Fill Valve	Ditto	Ditto	Ditto	None	No
896C	Accumulator #33 N_2 Fill Valve	Ditto	Ditto	Ditto	None	No

Valve	Function	Valve Position During Plant Operation	Assumed Consequences to ECCS Performance Due to Flooding of Valve Operator		Containment Isolation or Other Safety Related Functions	Design Change Proposed
			Short-term	Long-term		
896A	Accumulator #31 Drain Valve-WDS	Closed	None - No requirement for operation	None - No requirement for operation	None	No
896B	Accumulator #32 Drain Valve-WDS	Ditto	Ditto	Ditto	None	No
896C	Accumulator #33 Drain Valve-WDS	Ditto	Ditto	Ditto	None	No
896D	Accumulator #34 Drain Valve-WDS	Ditto	Ditto	Ditto	None	No
943	Accumulator Vent Valve	Ditto	Ditto	Ditto	None	No
955C	Accumulator #31 Sample Valve	Ditto	Ditto	Ditto	None	No
955D	Accumulator #32 Sample Valve	Ditto	Ditto	Ditto	None	No
955E	Accumulator #33 Sample Valve	Ditto	Ditto	Ditto	None	No
955F	Accumulator #34 Sample Valve	Ditto	Ditto	Ditto	None	No

Valve	Function	Valve Position During Plant Operation	Assumed Consequences to ECCS Performance Due to Flooding of Valve Operator		Containment Isolation or Other Safety	Design Change Proposed
			Short-term	Long-term		
1003A	R.C.D.T. Level Control Valve - W.D.S.	Open (Intermittent)	None - No requirement for operation	None - No requirement for operation	None	No
1003B	Ditto	Ditto	Ditto	Ditto	None	No
1163	R.C.F.C. Leakage Collection System Valve	Ditto	Ditto	Ditto	None	No
1164	Ditto	Ditto	Ditto	Ditto	None	No
1165	Ditto	Ditto	Ditto	Ditto	None	No
1166	Ditto	Ditto	Ditto	Ditto	None	No
1167	Ditto	Ditto	Ditto	Ditto	None	No
1609	P.R.T. Drain Valve - W.D.S.	* Ditto	Ditto	Ditto	None	No

SKETCH SHEET

CONCEPTUAL SKETCH
VALVE OPERATOR RELOCATION - SIS VALVES

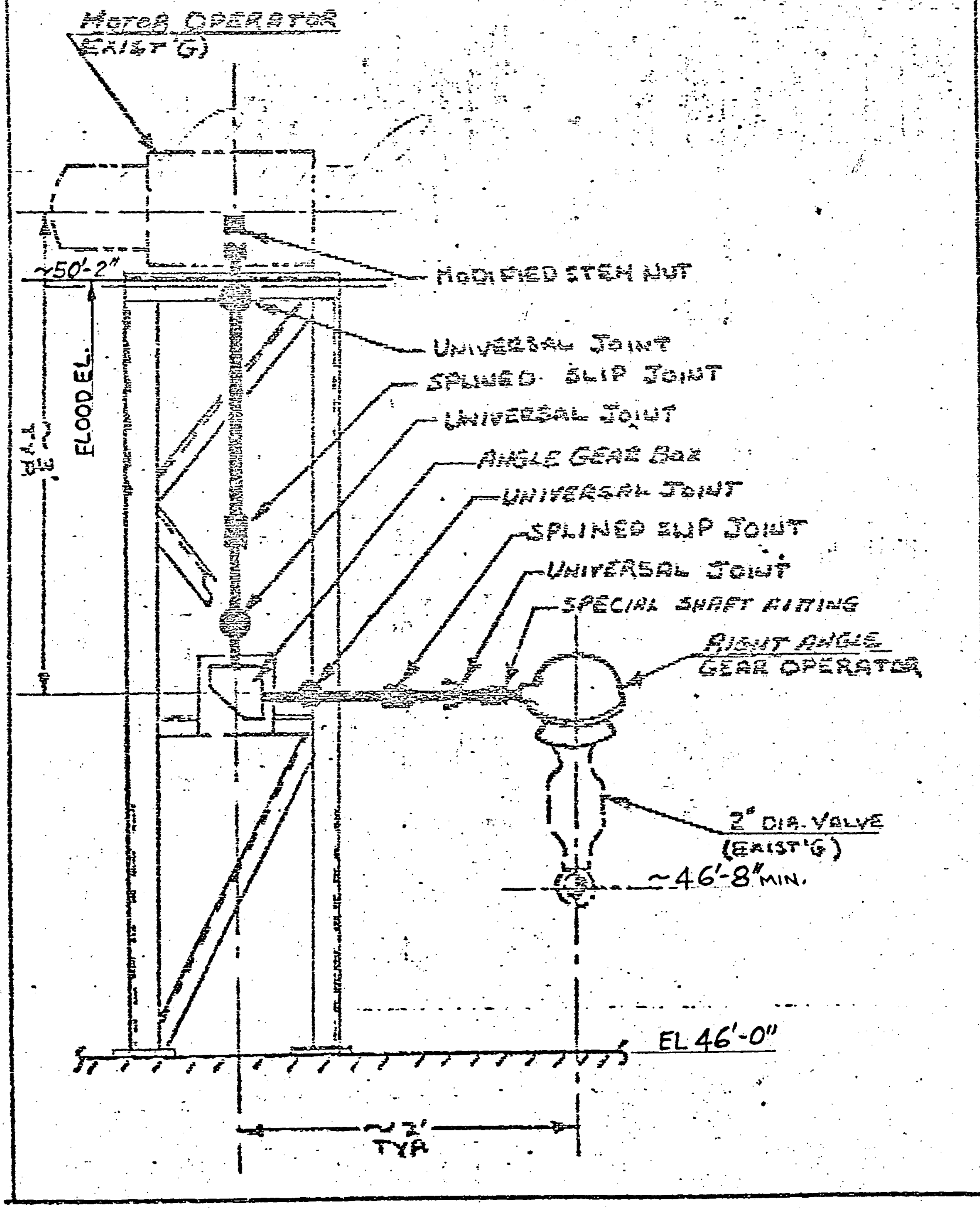
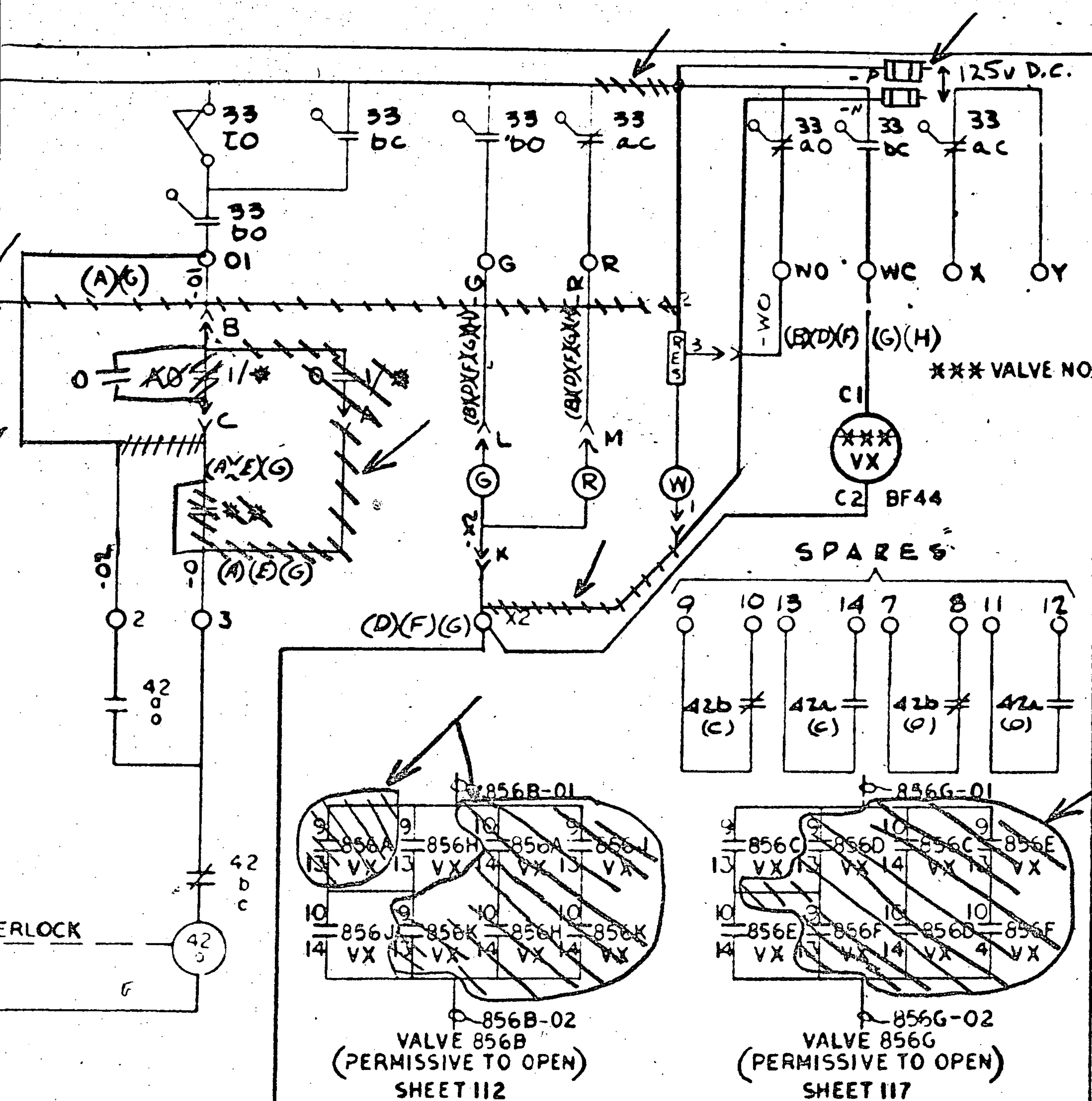


Figure 1

11/75



ARDS) MONITORLIGHT - OPEN

DC PNL 31 CKT 3	DC PNL 32 CKT 6
856C	856 H
856E	857J
856A	856 F
856K	856D

ECN-70058

VALVE SHOWN IN FULL OPEN POSITION

Westinghouse Electric Corporation F.P. 9321-05

TITLE CONSOLIDATED EDISON CO.-3307

INDIAN POINT STATION - UNIT No 3

ELEMENTARY WIRING DIAG. MOTOR OPERATED VALVE

S.O. 7AIT-305

P HEARN

500B971

SHEET - 155

1 SUB

ATOMIC POWER DIV.

PITTSBURGH, PA., U.S.A.

INT PROJECT

D FOR CONSTRUCTION

IPP-4336

F. DEVINE

50-4824

→ WORK CHANGES AS MARKED WITH REV2

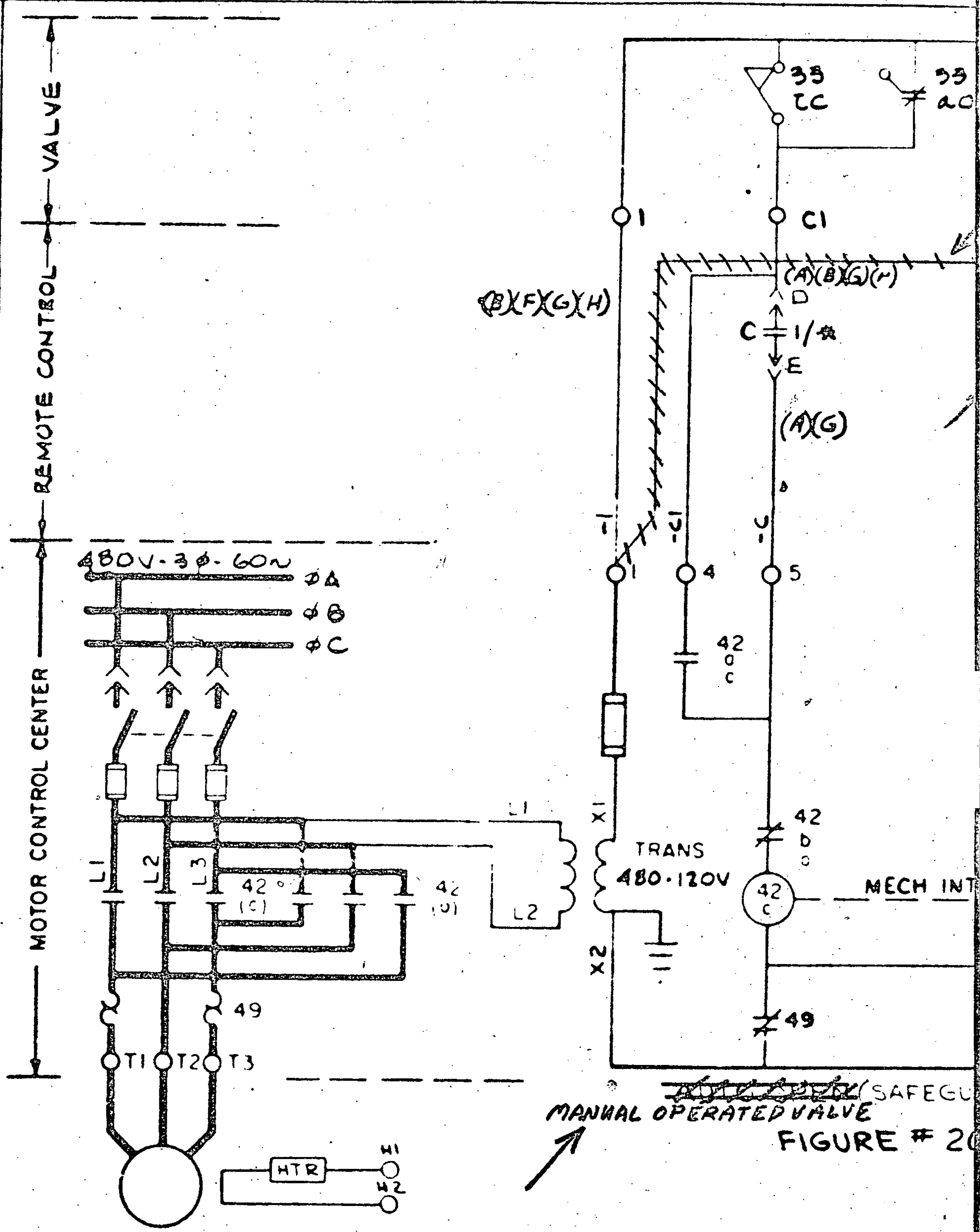


FIGURE # 2(

NOTE:

1. WIRE DESIGNATIONS ARE TO BE PREFIXED WITH VALVE NO.
2. LETTER (A) ETC. REFERS TO CABLE ~~SCHEMATIC~~

INDIAN P(

UNIT: 3

STATUS: CERTIFI(

CERTIFICATION: LETTER N(

AUTHORITY: R

ENGR. LTR. NO.: E

		125B			899B-C	R 11 SH.47
856G 36A-4FM	HIGH HEAD INJECTION LINE STOP VALVE	4/128	C/10	C/FAI	SEE SH. 155	SW. SB2F PNL IND. LGTS-SB2F PNL MON. LT. SB2F 33ac ANN SH 160 SEE NOTE 1
856H 36B-4FM	HIGH HEAD INJECTION LINE STOP VALVE	20/155 * 125A	C/10	O/FAI	856H-02 19 SI-21X 23 856H-O	SW./LGTS SB2F RELAY SB1R MON. LT. SB2F PNL 856H-VX ANN. SH. 160 S SEE NOTE 3 ~~113E303~~ ~~SH. 4~~ sh 112
856J 36B-8FM	HIGH HEAD INJECTION LINE STOP VALVE	20/155 * 125A	C/10	O/FAI	856J-02 18 SI-22X 22 856J-O	SW./LGTS SB2F RELAY SB1R MON. LT. SB2F PNL 856J-VX ANN SH. 160 S SEE NOTE 3 sh 112 ~~113E303~~ ~~SH. 4~~
856K 36A-8FM	HIGH HEAD INJECTION LINE STOP VALVE	20/155 * 125A	C/10	O/FAI	856K-02 19 SI-11X 23 856K-O	SW./LGTS SB2F RELAY SB2R MON. LT. SB2F PNL 856K-VX ANN SH. 160 S SEE NOTE 3 sh 112 ~~113E303~~ ~~SH. 4~~

OPER. MODE

O= NORMALLY OPERATED OPEN
C= NORMALLY OPERATED CLOSED
X= NORMALLY OPERATED OPEN & CLOSE
FC= FAILED CLOSED
FO= FAILED OPEN
FAI= FAILED AS IS

NOTE 1. SWITCH LEGEND PLATE TO BE ENGRAVED "CLOSE-OPEN"

2. REFERS TO CABLE SCHEM. 9321-LL-31263 SH.

ECN-70058

5	ECN-6887 N. MEARS 12/21/70 ... 1/4/71 ... 1/5/71
4	ECN-6811 WRS 10-26-70 ... 10-27-70 ... 30 70
3	ECN ... BACKCIRCLED 6-10-70 ...
2	ECN-... GERBER ... 4/17/70
1	
SUB	S.O. INT-385

Westinghouse Electric Corporation F.P. 9321-05-3387-8

TITLE: CONSOLIDATED EDISON INDIAN POINT STATION UNIT NO. 3

ELEMENTARY WIRING DIAG. VALVE TABLE-MOV

R.W. KEMERER ... 12/13/69

ATOMIC POWER DIV. PITTSBURGH, PA. U.S.A.

500B971

SHEET-117

WORK CHANGES AS MARKED WITH REV 12

INDIAN POINT PROJECT

UNIT: 3

STATUS: CERTIFIED FOR CONSTRUCTION

CERTIFICATION: LETTER NO.: IPP-4336

AUTHORITY: R. F. DEVINE

ENGR. LTR. NO.: E-PED-4824

VALVE (MCC LOC)	FUNCTION	FIG/SH	SW/SH	OPER MODE	** INTERLOCKS OR AUTO CONT. CONTACT	REF DWG & REMARKS
842 36A-7FM	MIN FLOW ISOLATION VALVE	8/132 * 127A	C/10	O/FAI	1R 842-C2 E5 ON 43/RS-5 F5 1S 842-C	SW & LGTS SBIF PNL MON LGTS SBIF 33ac ANN SH.160 33bc R LT SH.147
843 36B-7FM	MIN FLOW ISOLATION VALVE	8/132 * 127A	C/10	O/FAI	2A 843-C2 C11 ON 43/RS-5 D11 2B 843-C	SW & LGTS SBIF PNL MON LGTS. SBIF 33ac ANN SH.160 33bc R LT SH.147
743 36A-7RM	RESIDUAL HEAT REMOVAL LOOP OUTLET VALVE	8/132 * 127A	C/10	O/FAI	1T 743-C2 C1 ON 43/RS-5 D1 1A 743-C	SW & LGTS SNF PNL MON LGTS SBIF 33ac ANN SH.160 33bc R LT SH.147
1870 36B-7RM	RESIDUAL HEAT REMOVAL LOOP OUTLET VALVE	8/132 * 127A	C/10	O/FAI	2M 1870-C2 E7 ON 43/RS-5 F7 2N 1870-C	SW & LGTS SNF PNL MON LGTS SBIF 33ac ANN SH.160 33bc R LT SH.147
747C 36A-6RM	RESIDUAL HEAT LOOP DISCHARGE STOP VALVE	8/132 * 125B	C/10	O/FAI	747C C2 1F A7 ON 43/RS-5 B7 1H 747C C	SW.-SGF PNL IND. LGTS.-SGF & SBIF 33ac ANN. SH.160 MON. LGT.-SBIF 33bc "R" LT. SH.147
899B 36B-6RM	RESIDUAL HEAT LOOP DISCHARGE STOP VALVE	8/132	C/10	O/FAI	899B C2 2C G7 ON 43/RS-5 H7	SW.-(SGF) PNL. IND. LGTS.-SGF & SBIF 33ac ANN SH.160 MON. LGT.-SBIF

REV	
8	ECN-9557 CHANGES ARE CIRCLED P.H. FARM 10-25-72 11-1-72 11-1-72
7	ECN-7666 T.G. 5-16-71 6-23-71 6-23-71
6	ECN-7161 T.G. 3-9-71

		*125A					MON. LT. SB2F 33bc ANN. SH.160 SEE NOTE 1
856C 36A22RM	HIGH HEAD INJECTION LINE STOP VALVE ECN—70058	20/155 *125A	C/10	O/FAI	856C-02 4 SI-10X 8 856C-0		SW/LGTS SB2F RELAY SB2R SEE NOTE 3 MON. LT. SB2F 33bc ANN. SH.160 ~~113E303 SH 4~~
856D 36B2RM	HIGH HEAD INJECTION LINE STOP VALVE	20/155 *125A	C/10	O/FAI	856D-02 4 SI-20X 8 856D-0		SW/LGTS SB2F RELAY SB1R SEE NOTE 3 MON LT. SB2F 33bc ANN SH.160 ~~113E303 SH 4~~
866A 36A4FH	SPRAY PUMP DISCHARGE STOP VALVE	12/136 *127A	C/10	C/FAI	866A-C2 A1 W 43/RS1 B1 1L 866A-C3 A1 19 52b/CS1 B1 866A-C 866A-02 4 SI-1X 8 866A-0		SW.-SB1F PNL. IND. LTS.-SB1F PNL. MON LT. SB1F 33bc "R" LT SHT. 142 ~~113E303 SH 6~~

OPER. MODE

O = NORMALLY OPERATED OPEN
C = NORMALLY OPERATED CLOSED
X = NORMALLY OPERATED OPEN & CLOSED
FC: FAILED CLOSED
FO: FAILED OPEN
FAI = FAILED AS IS

NOTE:

1. SWITCH LEGEND PLATE TO BE ENGRAVED "CLOSE-OPEN"
2. * REFERS TO CABLE SCHEM 9321-LL-31263 SH.
3. NAMEPLATE TO READ "CLOSED OPEN"

REV	
5	ECN-6011 GERBER 10-8-70 10-27-70
4	ECN-6325 CHANGES BY KERBER PARKER
3	ECN-5011 GERBER 3-25-70
2	ECN 5522 R.W. KEMERER 12/1/69 12/4/69 12/15/69
SUB	S.O. INT-385

Westinghouse Electric Corporation
TITLE CONSOLIDATED EDISON CO.
INDIAN POINT STATION UNIT NO3
ELEM. WIRING DIA.
VALVE TABLE-MOV
R.W. KEMERER 4/13/69
ATOMIC POWER DIV.
PITTSBURGH, PA. U.S.A.
500B971
SHEET-112
F.P. 9321-C-3307

WORKED CHANGES AS MARKED WITH REV 13

UNIT: 3
STATUS: CERTIFIED FOR CONSTRUCTION
CERTIFICATION: LETTER NO.: IPP-4336
AUTHORITY: R. F. DEVINE
ENGR. LTR. NO.: E-PED-4824

INDIAN POINT PROJECT

SEE NOTE 3

VALVE Nº (MCC LOC)	FUNCTION	FIG/SH.	SW/SH	OPER MODE	** INTERLOCKS OR AUTO CONT. CONTACT	REF. DWG. & REMARKS
856E 36A-3RH	HIGH HEAD INJECTION LINE STOP VALVE ECN-70058	20/155 * 125A	C/10	O/FAI	12 16 856E-02 SI11X 856E-0	SW/LGTS SB2F RELAY SB2R MON LTS SB2F PNL 33C ANN SH 160 113E303 SH 4
856F 36B-8RH	HIGH HEAD INJECTION LINE STOP VALVE	4/155 *125A	C/10	O/FAI	12 16 856F-02 SI21X 856F-0	SW/LGTS SB2F RELAY SB1R MON LTS SB2F PNL 33C ANN. SH 160 113E303 SH 4
851A 36A-4RD	HIGH HEAD SIS PUMP Nº 2 STOP VALVE	14/138 * 138A	C/10	O/FAI	SEE SH. 138	SW. SB2F PNL IND LTS SB2F PNL MON. LTS SB2F PNL 33AC ANN SH 160 SEE NOTE 1
851B 36B-4RD	HIGH HEAD SIS PUMP Nº 2 STOP VALVE	15/139 * 138A	C/10	O/FAI	SEE SH. 139	SW - SB2F PNL IND LTS SB2F PNL MON LTS SB2F PNL 33AC ANN SH 160 SEE NOTE 1
856A 36A-2RH	HIGH HEAD INJECTION LINE STOP VALVE ECN-70058	20/155 * 125A	C/10	O/FAI	2 6 856A-02 SI-10X 856A-0	SW/LGTS SB2F RELAY SB2R SEE NOTE 3 MON LT SB2F 33AC ANN. SH 160 113E303 SH 4
856B 36B-2RH	HIGH HEAD INJECTION LINE STOP VALVE	4/128	C/10	C/FAI	SEE SH. 155	SW. - SB2F PNL IND LTS - SB2F PNL

11	ECN-9559 CHANGES ARE CIRCLED PHEARN 10-24-72 [illegible] 11-1-72 [illegible] 11-1-72
10	ECN-8182 J.G. 1/11/72 [illegible] 1-12-72 [illegible] 1-13-72
9	ECN-8351 J.G. 9-15-71 [illegible] 9-21-71 [illegible] 9-22-71
8	ECN-7665 J.G. 6-16-71 [illegible] 6-23-71 [illegible] 6-23-71
7	ECN-6887 W. MEARS 12/21/70 [illegible] 1/4/71 [illegible] 1/5/71
6	ECN-6651 J SHERWOOD 1-12 '70 [illegible] 12-8-70

1) Re-establish low head recirculation by opening RHR heat exchanger outlet valves MOV-746, MOV-747, MOV-889A and MOV-899B at the ACS panel.

2) Secure any running high head safety injection pumps.

3) Defeat the high head safety injection pump suction header low pressure alarm.

4) Isolate the low head - high head path by closing valves MOV-888A and MOV-888B.

2.3.J Hot Leg Recirculation

NOTE: Hot leg recirculation is required approximately 21 hours following the accident in order to assure that boric acid precipitation will not occur in the core. Recirculation is to be initiated as follows:

2.3.J.1 If containment spray and high head recirculation is not in effect-

a) Energize cold leg injection line motor operated valves MOV-856H and MOV-855-J and hot leg injection line motor operated valve MOV-856B at MCC-36B.

b) Close MOV-856J and MOV-856H <u>and</u> <u>open</u> MOV-856B.

NOTE: Valve 856B is provided with a permissive matrix which requires this valve sequence.

c) Energize cold leg injection line motor operated valves MOV-856C and MOV-856E and hot leg injection line valve MOV-856G at MCC-36A.

d) Close valves MOV-856C and MOV-856E <u>and</u> <u>open</u> MOV-856G.

NOTE: Valves 856G is provided with a permissive matrix which requires this valve sequence.

e) Shut Residual Heat exchanger discharge flow control valves HCV-638 and HCV-640.

f) Line up the low head to high head pass by opening valves MOV-888A and MOV-888B.

g) Open suction valves to No. 32 high head safety injection pump MOV-887A and MOV-887B.

h) Arm the high head safety injection pump suction header low pressure alarm by placing the manual toggle switch in the unblocked position. Observe the safety injection pump pressure indicator on the safeguard panel to ensure recirculated coolant is being supplied.

i) Start all three high head safety injection pumps.

2.3.J.2 If containment spray recirculation is in effect using low head to high head rath.

a) Shut the spray header isolation valves MOV-889A and MOV-889B.

b) Close MOV-856J and MOV-856H <u>and</u> <u>open</u> MOV-856B.

NOTE: Valve 856B is provided with a permissive matrix which requires this valve sequence.

c) Energize cold leg injection line motor operated valves MOV-856C and MOV-856B and hot leg injection line valve MOV-856G at MCC-36A.

d) Close valves MOV-856C and MOV-856E <u>and</u> <u>open</u> MOV-856G.

NOTE: Valve 856G is provided with a permissive matrix which requires this valve sequence.

e) Shut Residual Heat exchanger discharge flow control valves HCV-638 and HCV-640.

f) Open suction valves to NO. 32 high head safety injection pump MOV-887A and MOV-887B.

g) Start the third high head safety injection pump.

2.3.K During long term recirculation phase.

2.3.K.1 Closely monitor high head safety injection pump suction.

2.3.K.2 Low head recirculation may be established simultaneously with high head injection by opening the residual heat exchanger flow control valves HCV-638 and HCV-640. The total recirculation flow (sum of low and high head) must be limited to 3000 gpm per recirculation pump.

NOTE: This flow limitation is imposed to avoid the possibility of long term operation at or near cavitating conditions.

ENCLOSURE 2

Item 5 - Conformance with APPENDIX J to 10CFR50

(a) Air Lock Test After Each Opening

When closed, the air lock door seals are continuously pressurized by the weld channel system that is monitored at a pressure above peak accident pressure in accordance with Technical Specification requirements. A Control Room alarm is provided to alert the operators when the air pressure approaches the peak accident pressure.

When the air lock door is opened, a valve, operated by the door operating mechanism is closed to shut off the weld channel air to the door seals. When the door is subsequently closed, the door operating mechanism opens the isolation valve previously closed.

Plant personnel are assured that the air lock door seals are repressurized by audible indication (i.e. hearing the air being applied) and visual (control room indication of green door closed light). The mechanism that actuates the door closed limit switch also operates the valve that re-establishes weld channel air pressurization.

To assure that that valve body does not separate from the actuating mechanism, an extremely unlikely event, we propose to observe the pressure buildup on the air lock doors seals via the installed gages at each refueling outage. The gauges are observed on the containment side of each air lock door.

(b) Air Lock Test Every Six Months

It is proposed to test the containment air locks for unanticipated leakage paths at the same interval as the containment liner, with the initial test to be performed prior to core loading. The basis for this proposal is as follows:

Air lock weld seams, door seals, door operating mechanisms, and all other penetrations (except the two test connections on the outer bulkhead) are continuously sealed by the weld channel system to preclude leakage of containment atmosphere through anticipated leakage paths.

The air lock doors are designed such that containment pressure tends to seat the inner and the outer door tighter. Therefore, by applying pressure on the inside of the air lock, the inner door (into containment) tends to unseat causing air to leak by the seals.

A recent modification to the air lock design was accomplished to preclude a significant pressure buildup in the air lock. This modiifcation added two check valves in series that provide a vent path from inside the air lock to containment. Therefore pressurization of the air lock to perform the test will be done with the inner air lock door bolted up and the series check valves blocked.

We believe that the intent of the regulations is satisfied by these measures.

ENCLOSURE 2

Item 7 - <u>Plan for Utilization of Seismic Data</u>

The information below will be added to Supplement 32 to the response to FSAR Question 5.42.

5.0 <u>Comparison of Measured and Predicted Responses</u>

In the event the plant must be shut down based on the requirements of Items 4.1 or 4.3 above, the following analyses will be performed:

5.1 The time history records from the triaxial strong motion accelerograph on the containment base mat will be used to calculate response spectra at appropriate critical damping values. This response spectra will be compared with the spectra obtained from the peak shock recorder and the plant design response spectra.

5.2 Time history records from the triaxial strong motion accelerograph on the containment base mat will be used to determine the response of the Containment Building at elevation 100'-0" where the second triaxial strong motion accelerograph is installed. The calculated responses will be compared with the response directly measured.

5.3 The maximum acceleration recorded on the Peak Recording Accelerographs located on the steam generator, reactor coolant pump and the pressurizer will be compared with the calculated accelerations.

5.4 Should large discrepancies be noted between the actual response spectra and the design response spectra, and between actual building and equipment responses and calculated responses, the modeling and assumptions made for the dynamic analyses will be re-evaluated and reanalyses conducted as required.

ENCLOSURE 2

Item 8 - Radioactive Materials Safety

Section 11.3 of the FSAR will be modified in Supplement 32 to read as follows:

11.3 Radioactive Materials Safety

11.3.1 Materials Safety Program

Storage and handling of sealed sources will be in accordance with the regulations in Title 10 CFR Parts 20 and 70. The sealed sources are used to calibrate plant instrumentation and portable survey instruments at Indian Point Station. Control of the radioactive sources is maintained by the Radiation Safety Subsection. Licenses have been issued by the Atomic Energy Commission (AEC) for gamma and beta sources under the Facility Operating and By-Product Material License. The neutron source is licensed under a Special Nuclear Material (SNM) License along with the Facility Operating License.

When not in use, all radioactive sources not specifically part of an instrument, are kept in individual containers that provide shielding. The containers are lead pigs for the gamma emitting sources (small beta sources are not shielded) and a paraffin lined steel drum for the neutron source. The containers holding strong sources are kept, when not in use, in a locked source vault in a shielded room on the 53' elevation of the Unit No. 1 Nuclear Service Building. This room is constructed with one-foot thick concrete walls, floor and ceiling. The only entrance is a six-inch thick, hollow steel door, filled with lead shot. The door is normally locked and the key is issued by a Health Physics Supervisor or the licensed watch foreman. Containers holding weak sources are kept, when not in use, in a locked safe in the Chemistry Counting Room or the Radio Chem Lab also on the 53' elevation of the Unit No. 1 Nuclear Service Building. Both of these locations are in the controlled area.

Provisions are taken to protect personnel against undue exposure while handling radioactive materials and against undue contamination from bodily intake. This protection includes control by limitation of source use to qualified personnel with the permission of a Health Physics Supervisor. The sources will only be used under radiation area controlled conditions. Personnel will wear protective clothing, film or TLD whole body monitoring devices, and self-reading pocket dosimeters. Additional instrumentation will be used in the test area to monitor radiation. The source will be handled by the qualified personnel by means of a rod threaded to the tope of the source tongs or other remote handling devices.

Sources are handled in the Radio Chem Lab, the Chemistry Counting Room and at numerous radiation detector locations (PAB, air ejector, piping and electrical containment penetration area, the fan house and containment) of all three units.

The gamma sources will be transported from the locked source vault in their respective lead pigs to the calibration locations. The gamma sources are used to calibrate Area Radiation Monitors (ARM), Process Radiation Monitors (PRM), portable survey instruments and laboratory counters.

The neutron source will be removed from its container at the source vault, placed in a 1 or 2 liter polyethelene bottle filled with water and then transported to the calibration site. The neutron source is used to calibrate survey instruments and reactor flux instrumentation.

The beta sources are used to calibrate laboratory counters and survey instruments.

11.3.2 Facilities and Equipment

Exhaust hoods are located in the Controlled Area of each unit and are exhausted to that unit's containment vent system. The containment vent systems are monitored with continuous air monitors. The Radiation Safety Sub-section maintains numerous survey and measuring instruments including gamma, beta, alpha and neutron survey instruments as well as portable particulate and gas monitors.

11.3.3 Personnel and Procedures

Sources are handled by Health Physics Technicians, Chemistry Technicians, Health Physics Supervisors, and Chemistry Supervisors. Experience and qualifications of the key personnel responsible for handling and monitoring sealed sources is contained in the response to FSAR Question 12.3 and in Byproduct Material License No. 31-03112-01. Experience in handling sources has been acquired over a period of 3 to 12 years at Indian Point Station.

Radition safety instructions covering those operations are transmitted to working personnel in the form of Health Physics Procedures and Station Directives. All rules governing work in the Controlled Area, obtaining permission to enter the Controlled Area, wearing film or TLD badges and dosimeters, wearing proper anti "C" clothing, obtaining permission to use sources, use of source vaults and shielded containers, use of appropriate signs and barricades and the use of portable survey instrumentation are included in these instructions.

11.3.4 Required Materials

Isotope	Quantity Curies	Forms	Use
Pu-Be	1.0	Sealed	Calibration of reactor instrumentation and radiation monitoring equipment.
Cs-137	5.0	Sealed	Calibration
Cs-137	0.1	Sealed	Calibration

13027
50-286
ltr dtd 11-12-75

Regulatory Docket File

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Docket Clerk

WESTINGHOUSE DRAWING NO. 686J470

VIBRATION CHECK OUT FUNCTIONAL
TEST INSPECTION DATA (INT)

SECTION 1 OF 4
SECTION 2 OF 4
SECTION 3 OF 4
SECTION 4 OF 4

ENCLOSURE 2

Item 1. Onsite Geological Faults

See attached letter, (Cahill to DeYoung), dated November 10, 1975.