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REPORT ON THE MAY 2, 1979 TRANSIENT
AT THE OYSTER CREEK NUCLEAR GENERATING
STATION

BY

JERSEY CENTRAL POWER & LIGHT COMPANY

DATED: MAY 12, 1979

REPORT ON THE MAY 2, 1979 TRANSIENT AT THE
OYSTER CREEK NUCLEAR GENERATING STATION

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DESCRIPTION OF TRANSIENT AND SEQUENCE OF EVENTS RELATED TO SCRAM OF MAY 2, 1979, AT OYSTER CREEK NUCLEAR GENERATING STATION

INITIATING EVENT:

On May 2, 1979, at 1350 hours, an inadvertent reactor high pressure scram occurred during required surveillance testing on the isolation condenser high pressure initiation switches.

Two (2) sensors (RE-03A System I and RE-03B System II) (see Figure 1) of the four reactor high pressure scram sensors share a common sensing line with the isolation condenser high pressure initiation switches being tested.

The technician performing the test was in the process of verifying that the sensing line excess flow check valve V-130-1 was open when the scram occurred.

The scram has been attributed to a momentary simultaneous operation of switches RE-03A and RE-03B due to a hydraulic disturbance associated with valve manipulations required by procedure to verify the position of the excess flow check valve. The hydraulic disturbance also caused a momentary trip of the isolation condenser initiation switches (RE15A and RE15B). These sensors were not closed long enough to initiate an automatic initiation of the isolation condensers, since a time delay is involved in the initiation logic. However, these sensors also are used in the automatic recirculation pump trip logic which did operate in tripping the four operating recirculating pumps. No automatic time delay is involved in this logic.

INITIAL CONDITIONS:

Plant Parameters at the Time of the Scram:

Reactor Power	1895 MWt
Reactor Water Level (See Figure 2 for water level reference tabulation)	79" Yarway (13'-4" Above the top of the active fuel) 6.4' GEMAC

Reactor Pressure	1020 psig
Feedwater Flow	7.1×10^6 lbm/hr
Recirculation Flow	14.8×10^4 gpm

Equipment Out Of Service:

Relevent to Event Sequence:

A. One of the two (2) startup transformers, SB(Bank 6), was out of service as permitted by Technical Specifications, to perform an inspection of its associated 4160 Volt cabling. SB supplies offsite power to one half of the station electrical distribution system (see Figure 3) when power is not available through the station auxiliary transformer. The 4160 Volt buses which receive power from SB are 1B and 1D. Bus 1D supplies power to certain redundant safety systems. Bus 1D is designed to be powered from #2 Diesel Generator in the event power is not available from either the auxiliary transformer or startup transformer. Bus 1B supplies 4160 Volt power to non-safety related systems and hence, does not have a diesel backup power source.

B. One of the five (5) recirculation loops (D) was not in service due to a faulty seal cooler cooling coil. The pump suction valve was open, the discharge valve was closed, and the discharge valve bypass valve was open. No other systems and/or components important to the event sequence were out of service.

EVENT SEQUENCE: (To = 1350)

TIME OF EVENT (Sec)	EVENT DESCRIPTION
0	A reactor scram occurred for the reason previously described coupled with a simultaneous automatic trip of the four operating Recirculation Pumps. The Control

TIME OF EVENT (cont)	EVENT DESCRIPTION (cont)
	Room operator verified that all control rods inserted and proceeded to drive-in the IRM and SRM Nuclear Instrumentation. At this time, 4160 Volt power was being supplied from the auxiliary transformer during the coastdown of the Turbine Generating System and the Feedwater System was in operation. Recirculation flow started decreasing due to pump coastdown. Steam flow started decreasing due to loss of heat production (scram) but feed flow remained at the full power flow rate. Reactor vessel pressure decreased to the pressure regulator setpoint as steam flow decreased. Reactor water level began decreasing due to steam void collapse in the core.
13	The Turbine Generator tripped at the no load trip point which initiates an automatic transfer of power to the startup transformers. Power to Bus 1A and 1C successfully transferred from the auxiliary transformer to the SA (Bank 5) startup transformer. Since SB (Bank 6) was out of service at this time, power was lost to Buses 1B and 1D. As designed, Buses 1B and 1D separated through operation of breaker 1D and a fast start of Diesel Generator No. 2 occurred to power emergency loads on Bus 1D.

TIME OF EVENT (cont)	EVENT DESCRIPTION (cont)
	Loss of power to Bus 1B resulted in loss of Feedwater Pumps B and C and Condensate Pumps B and C. Although power was available to the A condensate and feedwater pumps, via Bus 1A, the A Feedwater Pump tripped on low suction pressure. Since water inventory was leaving the Reactor Vessel through the Steam Bypass Valves to the Main Condensers and a high capacity source of high pressure makeup water was not available, reactor water level and pressure decreased.
	In addition, the loss of power to Bus 1B caused the B Cleanup System Recirculation Pump to trip which, in turn, caused an isolation of the Cleanup System due to low flow through the cleanup filter. Furthermore, one condensate transfer pump and the operating fuel pool cooling pump tripped. An unsuccessful attempt was made to restart the A feedwater pump. (The reasons for the restart failure are described later.)
(Event Recorder) 13.6	Reactor water level decreased to the Low level scram setpoint which is 11'5" above the top of the active fuel region.
(Event Recorder) 16.8	The output breaker on the No. 2 Reactor Protection System M.G. Set tripped due to loss of power to the drive motor. The output voltage from the M.G. Set had

TIME OF EVENT (cont)	EVENT DESCRIPTION (cont)
	been maintained by flywheel action since the time of the turbine trip. Power to the M.G. Set drive motor is fed indirectly through Bus 1D which was deenergized at this time.
31	The No. 2 Diesel Generator Breaker closed and supplied power to the 1D Bus. A second control rod drive pump started.
43	Reactor water inventory continued to decrease due to steam flow to the main condenser. In anticipation of a Low Low Reactor Water Level automatic isolation of the reactor (which occurs at 7'2" above the top of the active fuel region), a manual reactor isolation was initiated to conserve inventory by closing the Main Steam Isolation Valves.
	This action was taken at an indicated water level of approximately 30" on the Yarway instrument which corresponds to 9'8" above the top of the active fuel region. It should be noted that the decrease in indicated water level and pressure was amplified by the effects of introducing cold feedwater into the vessel during the 13 second period prior to the Turbine Generator Trip. The cold feedwater reduced the steam voiding inside the vessel thereby causing a shrink in water level.

TIME OF EVENT (cont)	EVENT DESCRIPTION (cont)
49	The Main Steam Isolation Valves fully closed, thus stopping the loss of water inventory from the vessel thereby causing an increase in reactor steam pressure. Indicated reactor water level started to increase shortly after isolation, when reactor decay heat re-established a steam void distribution.
(Event Recorder) 59.6	The reactor mode switch was transferred from RUN to REFUEL.
76 (1 min. 16 sec.)	To establish a sink for the removal of decay heat from the reactor, the B isolation condenser was placed into service. At this time, the Control Room operator closed the A and E recirculation loop discharge valves (these valves take approximately two (2) minutes to close). It is postulated that at this time, both B and C loop discharge valves were also closed. The conclusion that the five recirculation pump discharge valves were closed is based upon loop temperature response later in the event and is further supported by the Low Low Low level at 172 seconds. The D loop was isolated previously. (See the equipment out of service section).
(Event Recorder) 90 (1 min. 30 sec.)	The reactor Low water level alarm cleared due to the water added from the isolation condenser to the Primary System.

TIME OF EVENT (cont)	EVENT DESCRIPTION (cont)
96 (1 min. 36 sec.)	The B isolation condenser initiation valve fully opened after 20 seconds. The temperature of the E recirculation loop, which serves as the B isolation condenser water return path, decreased due to the effects of cold water from the isolation condenser. The D recirculation loop temperature did not change appreciably. A, B, and C recirculation loop temperatures increased slightly. The heat-up is attributed to natural circulation through the partially open discharge valves carrying hot water (536°F) warming the lines previously cooled by the effects of cold feedwater. The reduced flow area between the lower downcomer and lower plenum area, due to the slow closure of the discharge valves, started to cause a shift in water inventory from the core area to the upper and lower downcomer region. The shift was due to the isolation condenser returning condensed steam from the core area to the downcomers. The water inventory shift continued as the discharge valves moved to the full closed position.
(Event Recorder)	
172 (2 min. 52 sec.) Last recorded point on the event recorder.	The reactor Low Low Low water level instrument trip point was reached. This was probably caused by the voided mixture in the separators having drained to the upper plenum, causing a reduction of static head above the Low Low Low water level instrument. This does not

TIME OF EVENT (cont)	EVENT DESCRIPTION (cont)
	necessarily indicate an inventory loss from the core but rather a redistribution of water and steam voids above the core.
186 sec. (3 min 6 sec)	All recirculation loop discharge valves fully closed. At this time, based upon closure initiation, the cooldown of the E recirculation loop stopped and a heat-up began. The indicated reactor water level increased due to the shift in water inventory. Recirculation loops A, B, and C continued to heat up. The mechanism of the heat up was due to heat transfer between the hot recirculation loop piping and the water in the piping. Reactor pressure continued to decrease as a result of isolation condenser operation.
250 (4 min 10 sec)	B isolation condenser was removed from service to reduce the rate of cooldown of the Primary System. Removal of the condenser caused indicated water level to decrease. The decrease in indicated water level was due to a return of water to the core region from the downcomer region through the five (5), two-inch (2") bypass valves around the recirculation loop discharge valves. During this period, the net water inventory effect was a storage of water in the recently secured isolation condenser. The recirculation loop discharge temperatures reached equilibrium and followed a slow cooldown trend.

TIME OF EVENT (cont)	EVENT DESCRIPTION (cont)
270 (4 min 30 sec)	The reactor pressure increased due to the effects of removing B isolation condenser. The rate of decrease in water level shifted from a ramp of approximately 37 in/min to 2 in/min. The reason for this change is the isolation condenser tube assembly was completely filled. The flow through the five (5) 2" bypass valves continued, accounting for the change in slope.
450 (7 min 30 sec)	Both isolation condensers were placed in service. This caused an increase in indicated water level and a decrease in pressure. The A recirculation loop temperature decreased because cold water from the A isolation condenser entered the A recirculation loop which is its return path to the reactor. A portion of the water passed through the loop via its 2" bypass valve, thus causing the cooldown.
528 (8 min 48 sec)	To slow the rate of cooldown, the B isolation condenser was removed from service. At this time, the indicated water level reached a maximum of approximately 14.4 feet above the top of the active fuel (88" on Yarway). This is considered to be above normal water level for full power operation. When the B isolation condenser was removed from service, indicated water level decreased to 13'8" above the top of the active fuel where it remained until approximately 1212 seconds when A

TIME OF EVENT (cont)	EVENT DESCRIPTION (cont)
	isolation condenser was removed from service. The reactor pressure continued to decrease and all recirculation loop temperatures continued to trend downward. Indicated water level was stable at this time because the head of water in the downcomer region was sufficient to establish equilibrium between the water entering the core region via the 5 two inch bypass valves and the condensed steam returning to the downcomer from the isolation condensers.
540 (approx) (9 min)	The four (4) Low Low Low water level indicators were verified locally to be below their alarm setpoint which is 10". The reading appeared to be at or below the instrument's lower level of detection.
810 (approx) (13 min 30 sec)	A recheck of the triple Low water level indicators showed that the pointers were active (moving) although they continued to read below their alarm point. The instrument was at or slightly above its lower level of detection.
1212 (20 min 12 sec)	A isolation condenser was removed from service, thus stopping the removal of inventory from the core region. Indicated water level decreased as the water in the downcomer region flowed into the core region. Reactor pressure started to increase due to the decay heat steam production.

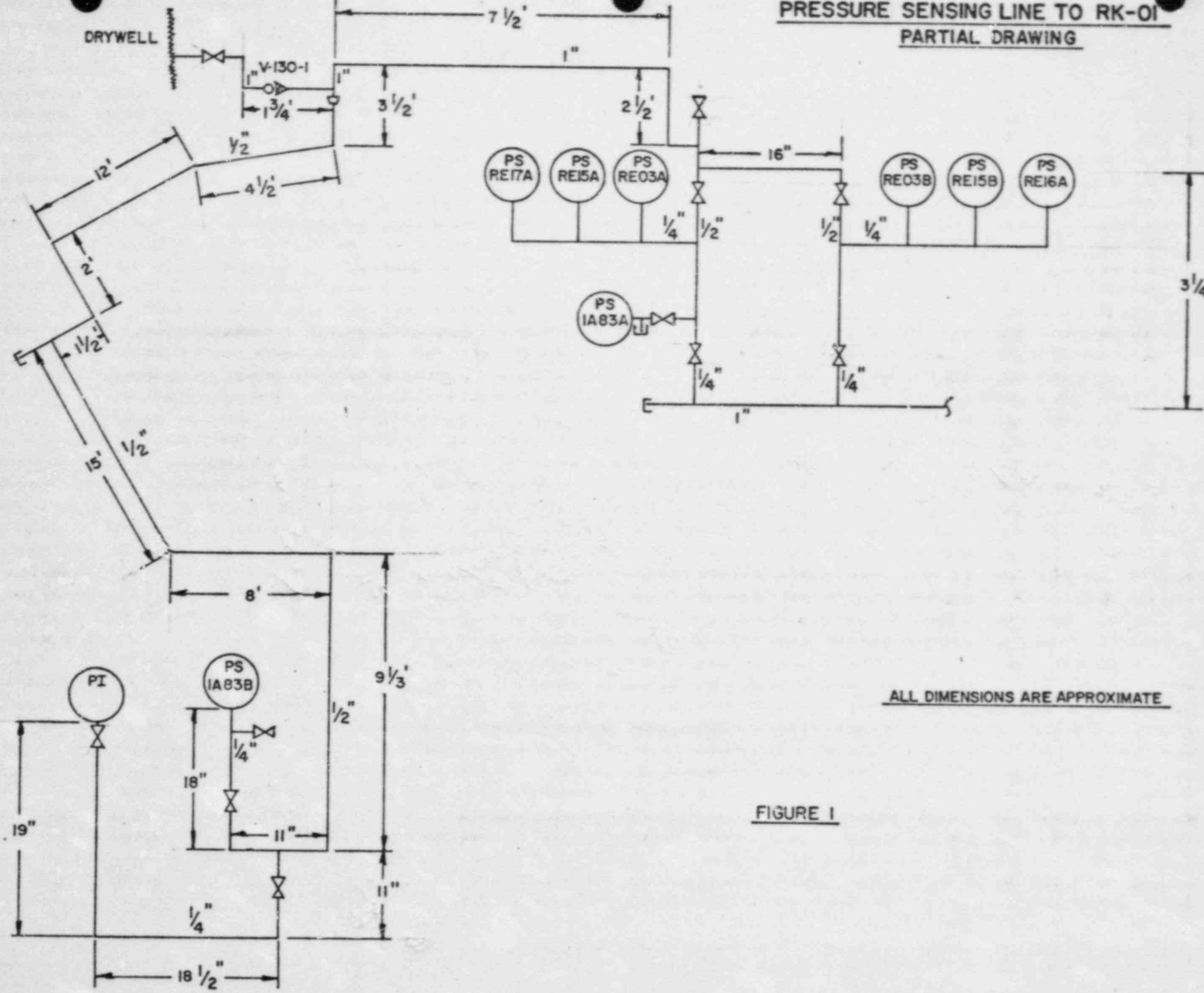
TIME OF EVENT (cont)	EVENT DESCRIPTION (cont)
1488 (24 min 48 sec)	The isolation condensers were used several more times to control the reactor cooldown with predictable increases in indicated water level and reduction in pressure. This mode of operation continued until 1914 seconds.
1914 (31 min 54 sec)	In order to more correctly determine the plant cooldown rate C recirculation pump was started and the discharge valve was opened. It was noted that the indicated water level dropped approximately 3 feet in less than 2 minutes. The C recirculation pump was shutdown and isolated to investigate the reason for the drop in level. In response to the indicated water level drop, additional attempt was made to start the A feedwater pump. The pump failed to start due to a tripped overload on the auxiliary oil pump which is interlocked in the pump starting sequence. The indicated water level started to increase due to the action of the operating isolation condenser transferring water to the downcomer region. When the C recirculation loop was started the loop temperature increased from approximately 400°F to 470°F. The other recirculation loop temperatures continued to trend down. At this time Low Low Low alarm may have cleared.

TIME OF EVENT (cont)	EVENT DESCRIPTION (cont)
2208 (36 min 48 sec)	The A Feedwater pump was successfully started by locally starting the auxiliary oil pump which satisfied the required starting interlocks. Indicated water level increased to a level corresponding to 13'8" above the top of the active fuel region. Realization occurred that the indicated water level and core water level may not have been the same when it was recognized that the five recirculation loop discharge valves were closed.
2340 (39 min 0 sec)	The A recirculation pump was placed in service at a flow rate of approximately 1.9×10^4 gpm, thus removing the disparity between water level measuring systems. The Low Low Low water level alarms were known to be cleared at this time. Indicated water level dropped approximately three feet to 11'4" above the top of the active fuel. The A recirculation loop temperature rose from 375°F to 465°F when it was placed in service. Steps were initiated at this time to bring the plant to "cold shutdown condition".
2700 (45 min 0 sec)	Reactor Protection System #2 restored and scram reset.
3600 (1 hr.)	The SB transformer was returned to service and Buss 1B was energized.

REACTOR PARAMETERS:

Figures 1-4a and 1-4b are a trace of reactor pressure, saturation temperature, annulus water level, recirculation flow, and recirculation loop temperatures from the time of the trip to 45 minutes later, when the transient was over. They are annotated with significant events during the period.

PRESSURE SENSING LINE TO RK-01
PARTIAL DRAWING



ALL DIMENSIONS ARE APPROXIMATE

FIGURE 1

	Building Elev.	Vessel Elev.	Dist. Above Active Fuel	Yarway (G+4)	GEMAC (Y-4)	[illegible]ton
Flange		660				
Steam Line		591				
Top of Steam Separations	84'8"	539				
Top of Indication			186"	100	8'	130"
Turbine Trip		529		90 RE05		
Hi Alarm Runout Reset		528	175"	89	ID14	
Normal		519		80	6.4'	
Lo Alarm		500		61	ID14	
Lo Level	80'7"	490	11'5"	51 RE05		
Bottom of Steam Separation		485				
Bottom of Dryer Skirt		477				
Feed Inst. "0"		443			0	
Lo Lo Level		439	7'2"	0 RE02		
Feed Line Penetration		422				
Lo Lo Lo Level		409	4'8"			0 RE18
Core Spray Nozzle		408				
Top of Active Fuel	69'2"	*353 5/16"	0			
Vessel "0"	39'9"	0				

*All other Levels rounded off to nearest inch

REF's: GE prints 104R858, 148F712, Burns & Roe prints 2063

FIGURE 2

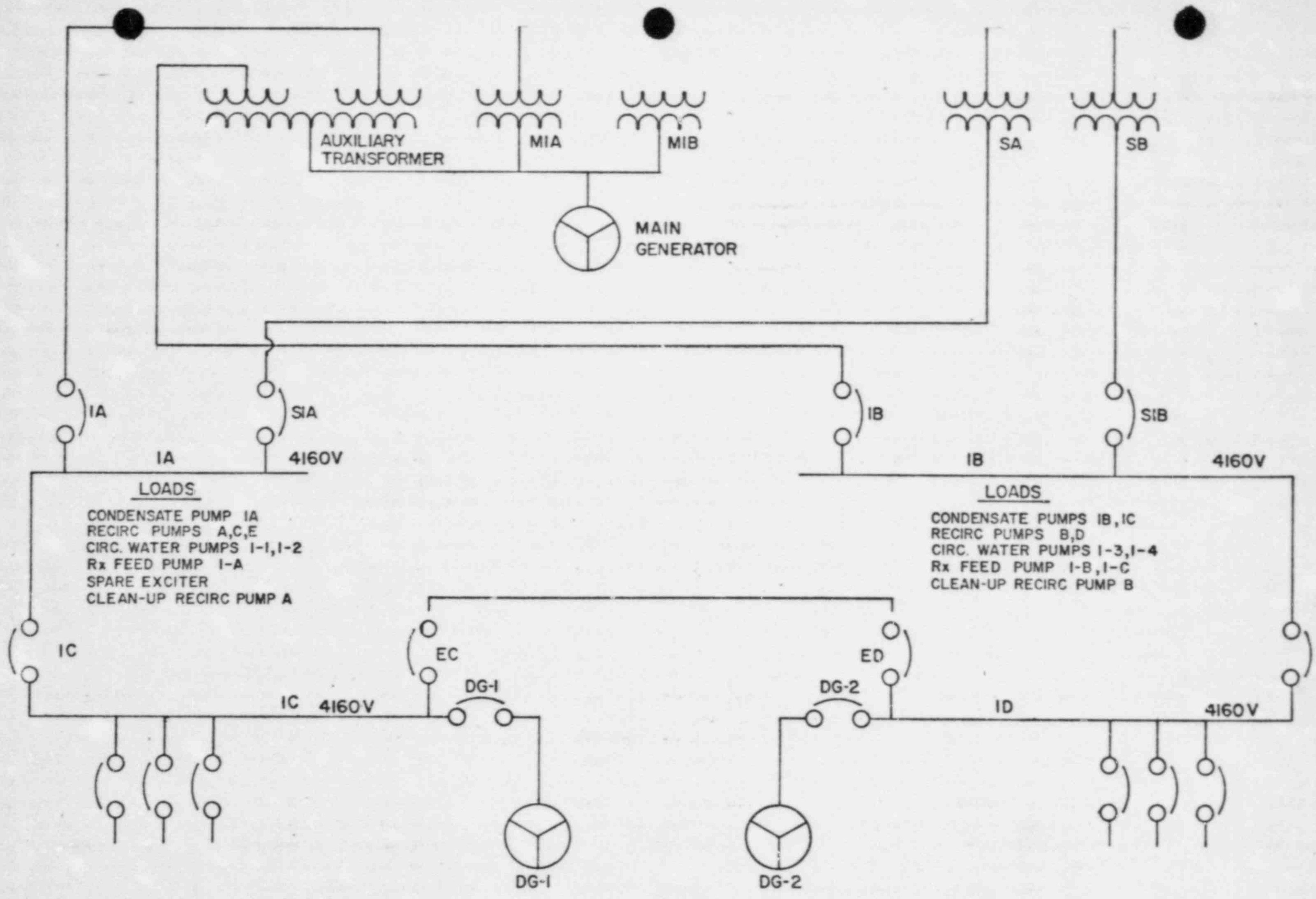

AUXILIARY TRANSFORMER
MIA
MIB
MAIN GENERATOR
SA
SB
IA
SIA
IB
SIB
IA
4160V
IB
4160V
LOADS
CONDENSATE PUMP IA
RECIRC PUMPS A,C,E
CIRC. WATER PUMPS I-1, I-2
Rx FEED PUMP I-A
SPARE EXCITER
CLEAN-UP RECIRC PUMP A
LOADS
CONDENSATE PUMPS IB, IC
RECIRC PUMPS B,D
CIRC. WATER PUMPS I-3, I-4
Rx FEED PUMP I-B, I-C
CLEAN-UP RECIRC PUMP B
IC
EC
ED
DG-1
DG-2
IC
4160V
ID
4160V
DG-1
DG-2

FIGURE 3

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DESCRIPTION OF THE OYSTER CREEK PLANT

The Oyster Creek Plant is a General Electric, 5 loop, forced recirculation, 1930 MWt, Boiling Water Reactor (BWR) with a Mark I containment system. The steam supply system consists of main steam piping, feedwater piping, and recirulcation pumps and piping. The system is also equipped with a cooling system consisting of circulation piping and condensers to provide for heat removal via natural circulation through the reactor. Various instrumentation and control systems are provided to monitor system performance and control operations. Figure 5 presents a simplified diagram of the above piping systems and their interconnections.

The containment system consists of a containment vessel (drywell) around the reactor vessel and recirculating system attached to a suppression chamber (torus). Steam released to the drywell is vented to the torus where the steam is condensed by the torus water which can be cooled by heat exchangers.

The main steam piping inside the drywell is equipped with 5 relief valves which can be operated either automatically or manually to relieve excess pressure or depressurize the system. Each of the 2 steam lines is also equipped with 2 isolation valves (1 inside and 1 outside of the containment vessel) to isolate the pressure vessel either automatically or manually as needed. The 5 relief valves operate automatically on high pressure to blowdown to the torus where the steam is condensed. These valves also actuate automatically when high drywell pressure reactor low-low-low water level, and core spray booster pump discharge pressure exists simultaneously for a period of 120 seconds or less. This is to

depressurize the system to allow for core spray flow into the pressure vessel. The main steam isolation valves are closed automatically on detection of any one of the following signals: (1) main steam line high radiation, (2) high steam flow in the main steam lines, (3) high temperature along the main steam lines, (4) main steam line low pressure, or (5) low-low reactor water level. These valves may also be closed manually by the operator.

The feedwater piping delivers feedwater through 2 check valves (1 inside and 1 outside of the containment vessel) and a locally operated stop valve, inside containment, to the feedwater sparger within the annular region (downcomer) of the reactor. This water mixes with the recirculation water in this region and is then delivered to the core through the recirculation loops.

The recirculation pumps take a suction from the annular region of the pressure vessel, between the vessel wall and the core shroud, through a normally open suction valve and discharge water through a discharge valve equipped with a 2" bypass valve into the bottom of the pressure vessel. The rated flow capacity of the combined recirculation loops is 61×10^6 lb/hr. Each recirculation loop is 26" diameter piping containing motor operated suction and discharge valves (equipped with 2" motor operated bypass valve) and a variable speed recirculation pump. There are 5 such recirculation loops and all suction, discharge and bypass valves are normally open during operation. Recirculation loops A and E have a 10" connection on the suction side of the recirculation pump upstream of the isolation valve. These connections are the return lines from the 2 isolation condensers.

The isolation condensers are connected to the reactor vessel steam region and the suction side of recirculation loops A and E providing a loop for natural circulation through the reactor core. The isolation condenser piping is 10" diameter piping with 2 isolation valves in the condenser inlet piping and 2 isolation valves in the condenser outlet piping. All valves are motor operated and normally open with the exception of the outside containment valve (DC motor operated valve) on the outlet piping which is normally closed. This system receives steam from the reactor vessel which is condensed within the tubes by surrounding water on the shell side and returns the condensed water to the recirculation loop. The heat transferred to the water on the shell side causes it to boil. The resulting steam is vented to atmosphere. The driving force for this system is natural circulation due to the heating of water in the core region. This system is actuated automatically on detection of a reactor high pressure or low-low water level after a maximum of 15 seconds time delay. The system may also be actuated manually by the operator.

Steam from the reactor drives the main turbine/generator, is then condensed and returned to the reactor via three 1/3 capacity condensate pumps, three 1/3 capacity feed pumps, and the feedwater piping. The 3 condensate pumps discharge to a common header feeding various heaters and coolers. Discharge water from the 3 intermediate pressure heaters feed the suction side of the 3 feedwater pumps which discharge through the 3 high pressure heaters into a common header feeding two 18" lines which run to a tee inside containment. Each of these lines then feed two 10" feedwater lines to the reactor.

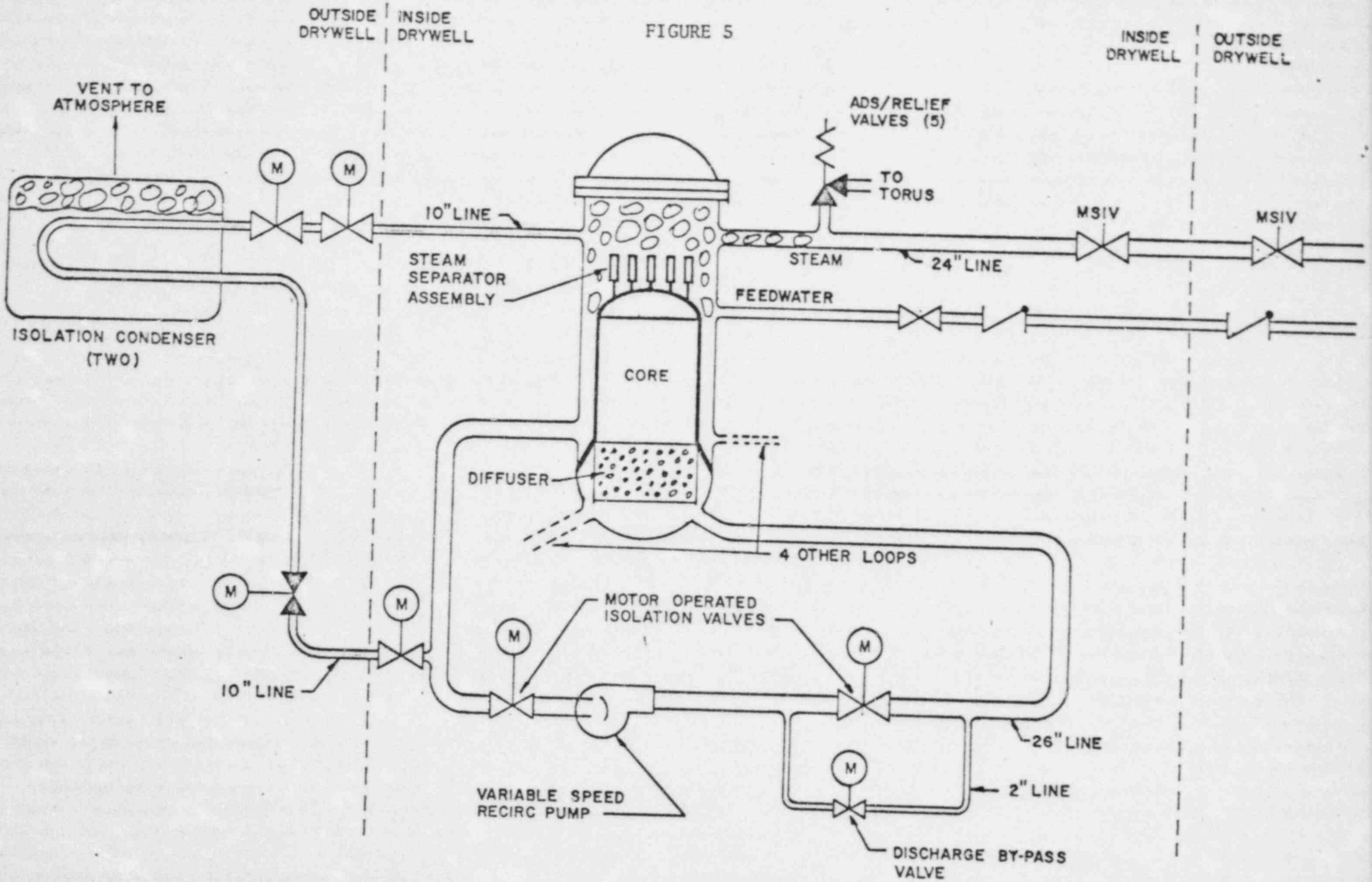
The condensate, feedwater, and recirculation pumps are powered from the station non-vital 4160 volt buses 1A and 1B which during operation receive power from the auxiliary transformers. Startup transformers SA and SB provide power to buses 1A and 1B during plant shutdown. Condensate pump 1A; recirculation pumps A, C, and E; feed pump 1A; and cleanup recirculation pump A receive power from bus 1A while condensate pumps 1B and 1C; recirculation pumps B and D; feed pumps 1B and 1C; and cleanup recirculation pump B receive power from bus 1B. Startup transformer SB provides power to bus 1A and startup transformer SB supplies power to bus 1B. Figure 3 presents a schematic diagram of this distribution system as well as the emergency power distribution.

In order to monitor system performance, instrumentation is provided to monitor reactor water level, reactor pressure, valve position, recirculation flow rate, and other system parameters. Reactor water level is monitored by three level measuring devices; "GE/MACS", "Yarways, and "Bartons". Two reference legs outside the vessel are provided for level indication and protective functions. Eight "Yarway" differential pressure cells, **four "Barton" differential pressure** cells, and three "GE/MAC" differential pressure electronic transmitters provide for level indication and protective functions. Reactor level is indicated both locally at instrument racks and remotely in the control room. Level indication in the control room is provided by both the "Yarway" and "GE/MAC" instruments. While the "Barton" instrument provides level indication at the instrument rack, it provides only an alarm function in the control room. All level indicator variable legs sense level in the annular region of the pressure vessel except for the 4 "Barton" Low-Low-Low level indicating switches. These switches indicate level inside the shroud

above the core. Local indication of this level is provided on an instrument rack with a remote alarm function in the control room. Figure 6 presents a diagram of the level indicators and associated alarm setpoints. Valve position indication is provided in the control room for the motor operated valves mentioned in the above discussion of systems.

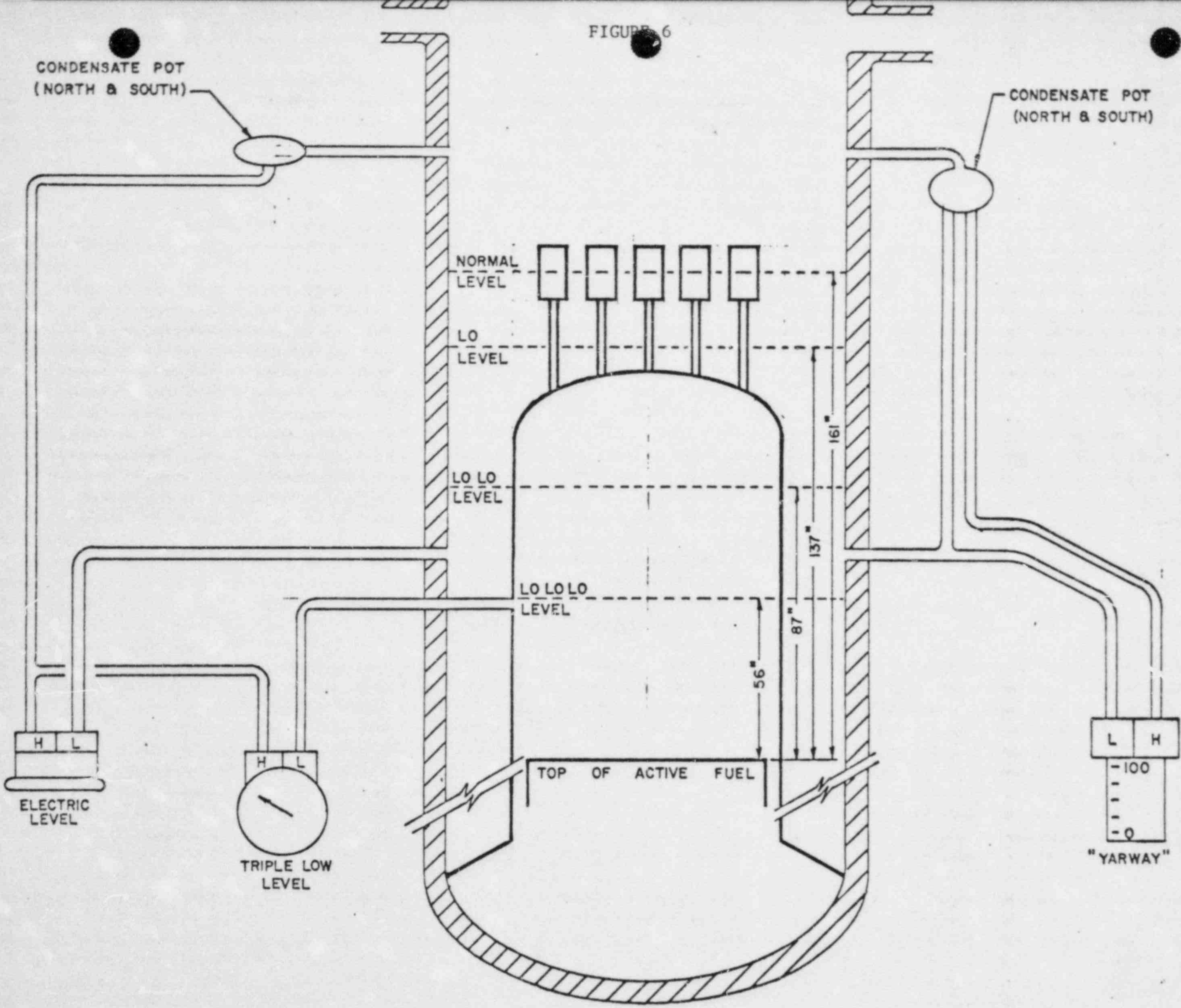
At the time of the Oyster Creek event all systems were in the normal line up with the exception of the startup transformer SB and recirculation loop D. Startup transformer SB was removed from service for maintenance. Recirculation pump D had been removed from the system due to a seal leak; therefore, the discharge valve was closed, suction valve open, discharge bypass valve open, and a plate was installed over the loop opening.

FIGURE 5



OUTSIDE DRYWELL
INSIDE DRYWELL
INSIDE DRYWELL
OUTSIDE DRYWELL
VENT TO ATMOSPHERE
ADS/RELIEF VALVES (5)
TO TORUS
M
10" LINE
MSIV
MSIV
STEAM SEPARATOR ASSEMBLY
STEAM
24" LINE
FEEDWATER
ISOLATION CONDENSER (TWO)
CORE
DIFFUSER
4 OTHER LOOPS
MOTOR OPERATED ISOLATION VALVES
10" LINE
26" LINE
2" LINE
VARIABLE SPEED RECIRC PUMP
DISCHARGE BY-PASS VALVE

FIGURE 6


CONDENSATE POT
(NORTH & SOUTH)
CONDENSATE POT
(NORTH & SOUTH)
NORMAL
LEVEL
LO
LEVEL
LO LO
LEVEL
LO LO LO
LEVEL
161"
137"
87"
56"
TOP OF ACTIVE FUEL
H L
ELECTRIC
LEVEL
H L
TRIPLE LOW
LEVEL
L H
100
0
"YARWAY"

2.A REACTOR WATER AND STACK GAS ANALYSES

A reactor water sample taken at 0820 on 5/2/79 showed all parameters to be within normal ranges.

The reactor scramed at 1350 on 5/2/79. A reactor water sample was taken at 1520 from the cleanup system inlet line after flow had been re-established through B loop. At that time "B" recirc. pump was off. The conductivity had risen from 0.10 μmho/cm in the morning sample to 0.37 μmho/cm which is normal after a scram. The gamma spectrum analysis showed the fission product activity concentrations to be in the normal expected ranges for the condition. Iodine 131 was up approximately a factor of 2 which is expected due to depressurization.

A reactor water sample was taken at 1640 from the cleanup inlet. "B" recirc. pump had been turned on prior to this sample. Comparison of the results with those of the previous sample showed good agreement. Conductivity had dropped to 0.30 μmho/cm. Gamma spectrum analysis showed normal isotopic decay and the effect of the cleanup demineralizer removal.

Reactor water samples were taken at 2105 from both the cleanup inlet and "A" recirc. loop. The samples showed good agreement on the parameters checked. The conductivity in the C.U. inlet sample was down to 0.18 μmhos/cm. Radioactivity levels continued to drop at normal rates.

A reactor water sample taken at 0641, on 5/3/79 showed all parameters within normal expected ranges for shutdown.

Four reactor water samples were taken on 5/3/79 at 0641, 0840, 1547, and 2000 hours. Isotopic analysis showed stable conditions in the water. When the sample taken on 5/4/79 @ 0800 continued the stable trend, the reactor water analysis was returned to the normal once per day frequency.

Additional backup analysis of the reactor water radioactivity concentration is being maintained with a continuous on-line gamma spectrum analysis. This multichannel analyzer has been in experimental operation as a part of an EPRI project at the Oyster Creek Site, since December 1978. After the plant shutdown, a continuous on-line analysis was started at 1611 hours and integrated over an eight hour period. The results of this analysis compare favorably with that of our grab samples taken during the period, and do not indicate any abnormal fission product levels in the water.

Data on a number of parameters measured in the primary coolant before and after shutdown are given in Tables 1 and 2. Similar data for three other scrams during the current cycle are presented in Tables 3-6 for comparison purposes. These tables also include data taken during subsequent startups so that expected levels during those periods are seen.

Stack Analysis

The stack particulate and charcoal filters were removed and analyzed on 5/3/79. The results were compared to the filters removed on 5/1/79 and showed no unusual or abnormal releases as the result of the plant shutdown. Data is presented in the attached Tables 7-10, where comparison is made also with previous scrams during this cycle. Data for previous scrams also includes stack releases during subsequent startups.

REACTOR WATER ANALYSES
Rx Scram at 1350
May 2, [illegible]79

TABLE 1

	0820 5-2-79	5-2-79 1520 Recirc Pump B Off	5-2-79 1640 Recirc Pump B on	5-2-79 2105 B	5-2-79 2105 A	5-3-79 0641 A	On Line Analysis see Note*5-2-79 start 1612
	μCi/ml						
I-131	7.47 E-3	1.53 E-2	1.34 E-2	2.61 E-3	4.07 E-3	1.36 E-3	Not detected
I-132	1.33 E-1	1.74 E-1	1.79 E-1	1.43 E-1	1.37 E-1	1.11 E-1	1.34 E-1
I-133	7.35 E-2	5.18 E-2	5.63 E-2	1.08 E-2	7.97 E-3	1.09 E-3	2.27 E-2
I-134	6.50 E-1	4.08 E-1	1.41 E-1	6.69 E-3	1.02 E-2	<1.0 E-4	-
I-135	1.64 E-1	1.10 E-1	6.95 E-2	2.10 E-2	9.27 E-3	<1.0 E-3	3.11 E-1
Xe-133	<1.34 E-3	5.19 E-2	1.29 E-2	1.19 E-2	8.40 E-3	2.19 E-2	-
Xe-135	2.35 E-2	3.79 E-2	1.33 E-2	1.49 E-2	4.75 E-3	2.49 E-2	5.22 E-2
Sr-91	2.80 E-2	2.40 E-1	7.01 E-2	1.67 E-2	1.19 E-2	1.58 E-3	3.10 E-1
Sr-92	1.00 E-1	4.05 E-1	1.02 E-1	8.79 E-3	5.34 E-3	<3.0 E-4	4.68 E-1
Mo-99	1.69 E-2	4.20 E-2	1.97 E-2	<1.00 E-2	1.56 E-2	<3.8 E-3	7.83 E-3
Tc-99m	8.91 E-2	7.00 E-2	5.56 E-2	4.11 E-2	4.64 E-2	3.66 E-2	8.30 E-3
Total Iodine	1.02 E 0	8.97 E-1	4.59 E-1	1.84 E-1	1.69 E-1	1.13 E-1	4.68 E-1
Np-239	1.35 E-2	1.47 E-1	9.71 E-2	1.48 E-1	1.45 E-1	6.50 E-2	1.36 E-2
Gross β	8.16 E-1					1.87 E-1	
Gross α	1.51 E-6					2.39 E-5	
pH	6.13			5.50	5.97	6.14	
Conductivity	0.10	0.37	0.30	0.18	0.25	0.15	
Sus. Solids	220 ppb						
C 1 -	<20 ppb						

*Continuous on line spectrum analysis integrated over 8 hour collection period starting at 1611 on 5-2-79

TABLE 2

REACTOR WATER ANALYSES

Parameter	5-3-79 0840	5-3-79 1547	5-3-79 2000	5-4-79 0845	5-5-79 0812	5-6-79 0752	5-7-79 0825
I 131	1.05 E-3	1.69 E-3	1.23 E-3	1.08 E-3	4.72 E-4	4.75 E-4	Not Det.
132	1.12 E-1	1.00 E-1	9.71 E-2	8.51 E-2	7.33 E-2	6.01 E-2	4.53 E-2
133	ND	ND	–	–	–	–	-
134	ND	ND	–	–	–	–	-
135	ND	ND	–	–	–	–	-
Xe 133	2.04 E-2	1.69 E-2	1.45 E-2	8.99 E-3	5.68 E-3	4.84 E-3	3.07 E-3
Xe 135	2.33 E-2	1.43 E-2	1.10 E-2	2.52 E-3	3.47 E-4	<3.3 E-4	<3 E-4
Tc 99m	4.53 E-2	4.70 E-2	4.49 E-2	3.87 E-2	2.98 E-2	2.24 E-2	1.68 E-2
Ba 140	6.38 E-3	<2.57 E-3	<2.56 E-3	3.72 E-2	4.11 E-3	2.36 E-3	<2.3 E-3
La 140	2.34 E-4	<2.14 E-4	<2.06 E-4	2.19 E-3	5.42 E-4	4.22 E-4	<2.0 E-4
Np 239	5.58 E-2	3.53 E-2	3.16 E-2	1.97 E-2	9.25 E-3	5.06 E-3	3.42 E-3
Gross Beta	1.59 E-1	–	1.43 E-1	1.05 E-1	6.84 E-2	6.34 E-2	4.26 E-2
Gross Alpha	<1.0 E-6	–	1.27 E-5	8.11 E-6	<1.0 E-6	<1.0 E-6	<1.0 E-6
pH	5.9	–	–	6.4	6.05	6.1	5.8
Conductivity	0.13	–	–	0.42	0.41	0.37	0.37

TABLE III
REACTOR WATER ANALYSIS - DEC. SCRAM 12-13-78 @ 1851
STARTUP 12-18-78 @ 1521

PARAMETER	8:20 12-13-78	0:30 12-14-78	8:30 12-14-78	8:30 12-15-78	8:20 12-16-78	8:30 12-17-78	8:30 12-18-78
	μCi/ml						
I-131	6.16 E-3	2.15 E-3	5.2 E-3	9.12 E-4	6.42 E-4	2.7 E-3 mpm	2.68 E-3
I-132	7.92 E-2	4.13 E-2	1.66 E-1	4.07 E-2	4.79 E-2	3.5 E-2	4.61 E-2
I-133	7.39 E-2	1.90 E-2	2.64 E-2	<4.77 E-4	1.35 E-2mpm	<3.18 E-4	<3.58 E-4
I-134	7.07 E-1	7.99 E-3	<1.91 E-3	<1.84 E-3	1.86 E-2mpm	1.51 E-2	<3.59 E-2mpm
I-135	1.68 E-1	1.54 E-2	1.15 E-2	<8.87 E-4	<5.91 E-4	4.58 E-4	<4.61 E-4
Xe-133	<6.49 E-3	6.57 E-3	<1.39 E-3	4.78 E-3	2.73 E-3	1.28 E-3	<3.58 E-4
Xe-135	2.40 E-2	1.40 E-3	1.85 E-3	5.16 E-3	4.20 E-4	1.92 E-4	<1.95 E-4
Sr-91	5.11 E-2	1.70 E-2	2.8 E-2	<1.61 E-3	<7.61 E-4	6.93 E-4	<6.88 E-4
Sr-92	1.53 E-1	1.25 E-2	3.59 E-3	<2.51 E-4	<1.95 E-4	1.67 E-4	<1.46 E-4
Mo-99	1.84 E-2	3.27 E-3	1.01 E-2mpm	<2.85 E-3	<1.59 E-3	1.50 E03	<1.56 E-3
Tc-99m	7.44 E-2	1.94 E-2	1.33 E-1	4.30 E-2	2.93 E-2	2.09 E-2	<2.96 E-2
Total Iodine	1.03 E-0	----	2.09 E-1	4.16 E-2	8.06 E-2	5.28 E-2	4.88 E-2
Np-239	4.61 E-2	3.79 E-2	2.9 E-1	3.83 E-2	9.91 E-3	6.24 E-3	1.0 E-2
Gross β	8.63 E-1		4.65 E-1	7.34 E-2	3.77 E-2	2.81 E-2	3.12 E-2
Gross α	ND		1.88 E-5	3.87 E-4	5.11 E-5	6.35 E-5	2.79 E-5
pH	6.9		6.2	6.3	6.25	6.55	5.94
Conductivity	0.2		0.15	0.35	0.27	0.4	0.50
Sus. Solids	230		<20	<20	260	140	<20
Cl^-	<20		34	<20	<20	<20	<20

REACTOR WATER ANALYSES - DEC. SCRAM CONT.

TABLE 4

PARAMETER	5:12 12-19-78	8:30 12-20-78	8:45 12-21-78				
	μCi/ml						
I-131	4.92 E-3	5.16 E-3	9.09 E-3				
I-132	5.19 E-2	6.28 E-2	9.46 E-2				
I-133	6.35 E-2	7.13 E-2	1.19 E-1				
I-134	4.75 E-1	6.31 E-1	8.1 E-1				
I-135	1.48 E-1	1.53 E-1	<3.6 E-3				
Xe-133	<8.34 E-4	1.06 E-3	<1.18 E-3				
Xe-135	1.84 E-2	3.03 E-2	2.65 E-2				
Sr-91	3.19 E-2	3.91 E-2	7.77 E-2 mpm				
Sr-92	1.22 E-1	1.39 E-1	1.82 E-1 mpm				
Mo-99	1.28 E-2	7.71 E-3	5.41 E-2				
Tc-99m	2.96 E-2	1.24 E-2	9.17 E-2				
Total Iodine	7.43 E-1	9.23 E-1	1.03 E+0				
Np-239	2.22 E-2	2.24 E-2	1.78 E-2				
Gross β	3.79 E-1	6.96 E-1	8.93 E-1				
Gross α	7.02 E-5	1.71 E-5	8.82 E-5				
pH	5.98	6.1	6.03				
Conductivity	0.33	0.15	0.15				
Sus. Solids	670	<20	110				
Cl^-	<20	31	30				

REACTOR WATER ANALYSES - JAN. SCRAM 1-15-79 @ 1552

STARTUP - 1-18-79 @ 1848

TABLE 5

PARAMETER	8:26 1-15-79	1:50 1-16-79	9:00 1-17-79	9:00 1-18-79	5:56 1-19-79	8:00 1-20-79	8:15 1-21-79
	μCi/ml						
I-131	1.31 E-2	5.99 E-3	2.69 E-3	2.28 E-3	1.34 E-3	6.71 E-3	6.65 E-3
I-132	2.34 E-1	1.73 E-1	1.69 E-1	<1.4 E-1	9.75 E-2	9.17 E-2	1.04 E-1
I-133	1.48 E-1	4.01 E-2	9.54 E-4	1.57 E-2	9.48 E-3	8.38 E-2	8.45 E-2
I-134	9.66 E-1	1.01 E-1	8.43 E-2	1.37 E-3	9.68 E-2	6.90 E-1	7.27 E-1
I-135	3.05 E-1	3.14 E-2	<1.09 E-3	<1.04 E-3	2.95 E-2	1.74 E-1	1.79 E-1
Xe-133	<2.11 E-3	4.76 E-3	<1.15 E-3	<9.84 E-4	<6.44 E-4	5.95 E-3	<1.36 E-3
Xe-135	3.51 E-2	9.98 E-3	<4.4 E-4	<4.24 E-4	4.32 E-3	1.89 E-2	2.61 E-1
Sr-91	6.78 E-2	5.19 E-2	<1.24 E-3	<1.3 E-3	5.8 E-3	4.36 E-2	5.4 E-2
Sr-92	1.78 E-1	1.28 E-2	3.37 E-2	6 E-1	2.13 E-2	1.67 E-1	1.71 E-1
Mo-99	1.70 E-2	4.82 E-3	3.56 E-3	8.82 E-3	8.71 E-3	1.71 E-2	1.06 E-2
Tc-99m	1.79 E-1	1.41 E-1	1.06 E-1	<1.25 E-3	3.24 E-2	4.82 E-2	7.43 E-2
Total Iodine	1.67 E-0	2.5 E-1	1.73 E-1	1.58 E-1	2.35 E-1	1.05 E-0	1.10 E-0
Np-239'	4.06 E-2	3.73 E-1	1.89 E-1	<2.73 E-3	2.95 E-2	1.95 E-2	2.33 E-2
Gross β	1.07 E-0	7.03 E-1	2.98 E-1	1.95 E-1	1.59 E-1	7.09 E-1	7.33 E-1
Gross α	ND	ND	2.69 E-5	3.95 E-6	2.72 E-5	2.65 E-5	ND
pH	6.25	6.15	6.0	6.3	6.91	6.2	6.25
Conductivity	0.145	0.175	0.2	0.14	0.16	0.12	0.112
Sus. Solids	<20	400	500	<20	330	<20	980
Cl^-	<20	36	<20	<20	43	<20	24

TABLE 6

REACTOR WATER ANALYSES FEB. - SCRAM 2-6-79 @ 1110

STARTUP 2-6-79 @ 2300

PARAMETER	8:46 2-5-79	8:23 2-6-79	4:55 2-7-79	9:00 2-8-79			
	μCi/ml						
I-131	6.5 E-3	6.8 E-3	3.5 E-3	6.85 E-3			
I-132	1.3 E-1	1.45 E-1	1.8 E-1	1.11 E-1			
I-133	8.0 E-2	8.17 e-2	2.6 E-2	8.19 E-2			
I-134	7.1 E-1	7.24 E-1	2.1 E-1	6.5 E-1			
I-135	1.8 E-1	1.78 E-1	6.9 E-2	1.63 E-1			
Xe-133	<1.33 E-3		2.7 E-2	2.11 E-3			
Xe-135	1.81 E-2		8.7 E-3	2.03 E-2			
Sr-91	3.69 E-2		1.3 E-2	3.3 E-2			
Sr-92	1.2 E-1		5.2 E-2	1.2 E-1			
Mo-99	2.0 E-2		1.6 E-2	1.0 E-2			
Tc-99m	1.0 E-1		6.9 E-2	7.8 E-2			
Total Iodine	1.11 E-0	1.14 E-0	4.92 E-1	1.01 E-0			
Np-239	2.45 E-2		1.35 E-1	3.4 E-2			
Gross β	8.35 E-1	9.09 E-1	4.04 E-1	8.03 E-1			
Gross α	ND	1.51 E-6	1.17 E-5	1.48 E-6			
pH	6.05	6.6	6.1	5.8			
Conductivity	0.145	0.122	0.2	0.14			
Sus. Solids	40	280	170	<20			
Cl^-	50	<20	40	<20			

STACK DATA - DECEMBER SCRAM: Scram 12-13-78 @ 1851
Startup 12-18-78 @ 1521

		Start	Stop	Release Rates	
Stack Sample	#S-101-78		9:30 - 12/11/78	I-131	0.41 µCi/sec
				I-133	1.25
				I-135	17.5 (.46 µCi/sec)
				Total Tech. Spec % 11.5%	
	#S-102-78	9:30 - 12/11/78	8:30 - 12/15/78	I-131	0.34
				I-133	1.59 (.48 µCi/sec)
				I-135	.22
				Total Tech Spec % 11.94%	
	#S-103-78	8:30 - 12/15/78	8:17 - 12/19/78	I-131	0.096
				I-133	0.102 (.100 µci/sec
				I-135	0.489
				Total Tech Spec % 2.5%	
	#S-104-78	8:17 - 12/19/78	8:10 - 12/21/78	I-131	0.43
				I-133	0.94 (0.59 µCi/sec)
				I-135	2.43
				Total Tech Spec % 14.8%	

TABLE 7

Scram Jan. 15 @ 1552
Startup Jan. 18 @ 1848

Jan. 8, 1979 - Jan 12 -	I^{131}	.23 µCi/sec	
	I^{133}	.75 µCi/sec	(0.34 µCi/sec)
S-5-79	I^{135}	.72 µCi/sec	
		Total Tech Spec	8.58%
Jan. 12, 1979 - Jan. 16	I^{131}	.39 µCi/sec	
	I^{133}	.80 "	
S-6-79	I^{135}	.50	(0.51 µCi/sec)
Scram 1-15-79		Total Tech Spec	12.75%
Jan. 16, 1979 - Jan. 19	I^{131}	.19 µCi/sec	
	I^{133}	.07 µCi/sec	(0.19 µCi/sec)
S-7-79	I^{135}	.01 µCi/sec	
on line 1-19-79		Total Tech Spec %	4.7%
Jan. 19, 1979 - Jan. 22.	I^{131}	1.04	
		1.33	(1.09 µCi/sec)
S-8-79		1.41	
		Total Tech Spec %	27.35%

TABLE 8

February 1979 Scram Stack Data   Scram 2-6-79 @ 1110
Startup 3-6-79 @ ~2300

S-12-79 1-30-79 to 2-2-79	I^{131} I^{133} I^{135}	0.45 μCi/sec 0.98 μCi/sec 0.78 μCi/sec % Tech Spec Total 11.5%	(.46 μCi/sec)
S-13-79 2-2-79 to 2-6-79	I^{131} I^{133} I^{135}	0.24 μCi/sec 0.78 μCi/sec 0.83 μCi/sec % Tech Spec 6.3%	(0.25 μCi/sec)
S-14-79 2-6-79 to 2-9-79 Scram 2-6-79 on line 2-7-79	I^{131} I^{133} I^{135} % Tech Spec 8.26%	0.32 1.12 1.17	(0.33 μCi/sec)
S-15-79 2-9-79 to 2-11-79	I^{131} I^{133} I^{135} % Tech Spec 10.76%	0.48 0.81 0.91	(0.43 μCi/sec)

TABLE 9

May Scram - Stack Data   Scram 5-2-79 @ 1350

S-40-79 to 5-1-79	I^{131}	0.241	
	I^{133}	1.59	(0.26 μCi/sec)
	I^{135}	2.05	
	% Tech Spec 6.5%		
S-41-79 5-1-79 to 5-3-79 scram 5-2-79	I^{131}	0.32 μCi/sec	
	I^{133}	0.85 μCi/sec	
	I^{135}	0.47 μCi/sec	
		% Tech Spec 8.3%	
S-42-79 5-3-79 to 5-7-79	I^{131}	0.0858	
	I^{133}	0.0218	(0.087)
	I^{135}	Not detected	
		% Tech Spec 2.2%	

TABLE 10

2.B. CORE WATER LEVEL ANALYSES

The previous section described the extensive activity surveys which were conducted to determine that no fuel damage had occurred as a result of the event. No activity indications above normal levels were observed. This section outlines the analyses and results which were completed to demonstrate that no loss in fuel integrity would have been expected as a consequence of the scram event.

Because of the initiating scram, the power drops rapidly through the fission power decay to normal decay heat levels. Even with the simultaneous recirculation pump trip, the pump coastdown retains an increasing flow to power ratio until natural circulation flow is established. Thus, the critical power ratio stays above the operating level prior to the scram.

The decay heat drops to sufficiently low levels after the first few seconds, that following recirculation pump coastdown the fuel may be cooled sufficiently in a pool boiling mode, provided a two-phase mixture level is maintained above the core. Therefore, the objective of the analyses is to determine the minimum mixture level above the active fuel region.

There are two basic evaluations which may be performed to determine the core mixture level. The first method is to conduct a mass and energy balance on the core accounting for the boiloff rate as a function of decay heat and saturation conditions, balanced against makeup from natural circulation flow and control rod drive flow into the lower plenum. The second approach is to utilize a mass inventory allocation process to distribute the available mass through the system depending upon known volumes along with known levels or known thermodynamic conditions.

Both of the above approaches make use of the fact that the system is isolated by MSIV closure and the mass at that time is retained and further augmented by Control Rod Drive flow. In order to determine the total amount of mass in the system, the time of low-low-low water level indication (172 seconds) is utilized as an initialization point. At that time, water levels in both the downcomer and core are known, one isolation condenser is in service, loop temperatures are known and the total system mass may be estimated. Subsequent changes in the mass allocation is utilized to determine the core mixture level.

Analyses of the minimum water level were calculated by both Exxon Nuclear Company and General Electric Company. The Exxon evaluation included both the boiloff approach and the mass allocation technique. The General Electric analysis determined the water level by the boiloff process. Details of the GE analysis are shown in Appendix 1.

A summary of the results of the various analyses are shown in Table 2.B.1. As can be seen, each evaluation concluded that the core remained adequately covered by a two-phase mixture level throughout the course of the event. Therefore, no loss of fuel integrity would have been expected as a consequence of this event and indeed no indications of fuel failure were observed.

TABLE 2.B.1

RESULTS OF WATER LEVEL ANALYSES
OYSTER CREEK SCRAM OF MAY 2, 1979

	Exxon Analyses*		GE Analyses
Initial Mass Inventory Above Core	7.12 ft. (two-phase mixture)		31,600 lbs.
Type of Analysis	Inventory	Boiloff	Boiloff
Minimum Mixture Level Above Top of Active Fuel	1.62 ft.	2 ft.	2.38 ft.
Time of Minimum Mixture Level	29 min.	20 min.	12 min.

*Preliminary results, final report in process.

3.A PROCEDURAL CHANGES

Operator action specified by Procedure 501 in response to the VESSEL LEVEL TRIPLE LOW annunciator has been revised to include lessons learned from the Oyster Creek scram of May 2, 1979.

Plant procedures affecting reactor recirculation pumps have been reviewed. The procedure review included appraising whether the procedure had been adequately revised to reflect the recirculation pump trip modification and if instructions in the procedures could have contributed to the incident under investigation.

Standing Order #23 related to the operation of the isolation condensers has been deleted. The required actions necessary for manual initiation of the isolation condensers have been incorporated into plant Procedure 307.

All Standing Orders have been reviewed to ensure that they do not impede or preclude the automatic operation of an engineered safeguard system.

The procedures reviewed as a result of IE Bulletin 79-08 and the Oyster Creek scram of May 2, 1979, are listed below. The considerations that were reviewed against are noted. Procedures for which a change request has been submitted are marked by an asterisk.

NOTES	PROCEDURE NOS. AND TITLES
	*204.1 - Scram Recovery
	e*301 - Nuclear Steam Supply System
b,c	*307 - Isolation Condenser System
b,c	*317 - Feedwater System
b,c	*318 - Main Steam System and Reheat System
a,b,c,	e*501 - Annunciators and Alarms
a	*502.1 - Loss of 230 KV Lines
a	502.3 - Loss of 4160 Volt Bus 1A (1B, 1C, 1D)
a	502.4 - Loss of the Reactor Protection System Power
a	502.5 - Loss of 125 V D.C. Power
a	*502.6 - Complete Loss of AC Power
a	*503 - Instrument Air Failure
a	504 - Service Water Failure
a, c	505.1 - Recirculation Flow Increase
a,b,c	*505.2 - Recirculation Flow Decrease
a	506.1 - Rod Drop
a	506.2 - Loss of CRD Hydraulic System
a	506.3 - Abnormal Control Rod Motion
a	506.4 - Rod-to-Drive Coupling Failure
a,b,c,	e*506.5 - Scram System Failure
a, c	507.1 - Reactor Building Closed Cooling Water System Failure
a, c	*507.2 - Turbine Building Closed Cooling Water System Failure
a	508 - Loss of Vacuum
a	509 - Inadvertent Opening of Turbine Bypass Valve(s)

a *510 - Turbine Trip

a,b,c,d *511.1 - Feedwater Pump Failure

a *511.2 - Condensate/Feedwater System Rupture

a 511.3 - Feedwater Flow Control Failure

a 511.4 - Loss of Feedwater Heaters

a 512.1 - Loss of Generator Excitation

a 512.2 - Generator Excitation Equipment Malfunction

a 512.3 - Loss of Generator Stator Cooling

a 513 - Generator Trip

a,b,c,d,e *514 - Reactor Isolation Scram

a 515.1 - Small Piping Leaks in the Turbine Building

a 515.2 - Small Piping Leaks in Reactor Building

a *515.3 - Small Piping Leaks in Drywell

a *516.1 - Main Steam Line Rupture Outside Drywell

a *516.2 - Piping Rupture Inside Drywell, Offsite Power Available

a *516.3 - Piping Rupture Inside Drywell with Loss of Offsite Power

a *516.4 - Isolation Condenser Line Break Outside Drywell

a *516.5 - Piping Rupture Inside Drywell with Loss of Offsite Power and One Diesel Generator Inoperable

a *517 - Significant Increase in Off Gas Release Rate

a 518 - Inadvertent Liquid Release to Discharge Canal

a *519 - Loss of Containment Integrity

a 520 - Hurricane

a 521 - Hazardous Condition on the Refueling Floor

a 522 - Inadvertent Poison Injection

a 523 - Condenser Tube Leakage

a 524 - Cleanup Filter Cake or Demineralizer Resin Breakthrough

a *525 - Loss of Drywell Cooling

a 526.1 - Fire in Plant Areas Other Than Control Room

a *526.2 - Fire in the Control Room

a *527.1 - Inadvertent Relief Valve Actuation While at Power

a *527.2 - Failure of Relief Valve to Reseat - Reactor Scrammed

a *527.3 - Loss of Feedwater - Electromatic Relief Valve Failure

a 528 - Tornado

a *529 - Emergency Containment Purge

a 530 - Loss of the Reactor Shutdown Cooling System

a 531.1 - Loss of SRM Instrumentation

a 531.2 - Loss of IRM Instrumentation

a 531.3 - Loss of APRM Instrumentation

a,b,c,d,e *532 - Automatic and Manual Reactor Scram

a,b,c *534 - Loss of Reactor Cooling Mechanisms During Reactor Shutdown

a 535 - Inadvertent Reactor Criticality

a 536.1 - AOG System

a 536.2 - Radwaste Service Water Failure

a 536.3 - AOG Closed Cooling Water Failure

a 536.4 - Off Gas Building Loss of Power

a 536.5 - Fire in AOG Charcoal Bed

a 537.1 - Radwaste Building Closed Cooling Water Failure

a 538 - Off-Gas Explosion

a 539 - Response to Malfunction of Meteorological Instrumentation

e*603.3.001 - Recirculation Pumps Trip Circuitry Test

*604.4.013 - Pressure Suppression Chamber (Torus) External Inspection

e*609.3.003 - Isolation Condenser Automatic Actuation Sensor Calibration & Test

e*619.3.004 - Reactor Lo Lo Water Level Functional Test

e*636.2.001 - Diesel Generator Automatic Actuation Test

NOTES:

a. Review Considerations:

1. Are the operators directed to override automatic action of safety systems except when continued operation would result in unsafe plant conditions?

2. Is primary containment isolation on automatic initiation of core spray prevented or hindered by this procedure?

3. Is inadvertant, undesired pumping, venting, or release of radioactive liquids or gasses from primary containment possible per this procedure?

4. Is unrestricted resetting of isolation signals allowed by this procedure?

5. Is the operator given parameters other than reactor level to determine plant status?

b. Review Considerations:

1. Does this procedure contain instructions which could be interpreted to direct closure of recirculation loop isolation valves in all loops?

c. Review Considerations:

1. Does this procedure require additional caution statements to adequately warn the operator against closure of recirculation loop isolation valves in all loops?

d. Review Considerations:

1. Does this procedure require additional guidance to adequately direct the operator in the event of a total loss of feed and 5-recirc pump trip?

e. Review Considerations:

1. Does this procedure require revision to adequately reflect the recirculation pump trip modification?

2. Did the instructions in this procedure contribute to the May 2, 1979, Oyster Creek incident?

3. These procedures have been reviewed to ensure that adequate guidance to operators on mode switch changes following a transient have been provided.

3.B OPERATOR TRAINING

The training planned as a result of the Oyster Creek incident of May 2, 1979, includes both short-term and long-term training. Short-term training has two parts. The first part is a review session with all licensed Control Room Operators and their Group Supervisors to cover the sequence of events, lessons learned, and changes to procedures, equipment, and policy anticipated as a result of the incident. The second part is a detailed review of the specific operating and emergency procedure changes resulting from the incident. Long-term training will place additional emphasis on active participation by the Control Room Operators, and especially their Group Supervisors, in efforts to better prepare them for unexpected events. The training methods include problem solving, simulated drills, and abnormality response sessions. To address conflicting indications, the operators will receive training to help them better analyze plant conditions, verify the most conservative indication, and act accordingly.

The incident will be reviewed with each licensed operator prior to shift relief.

The procedure review will be completed with each shift during the regularly scheduled training week following the approval of the procedure changes.

The long-term training is planned to include a continuing emphasis on abnormality response.

3.C SURVEILLANCE CHANGES

The following steps and actions have been taken to minimize the likelihood of an instrument surveillance test initiating a similar event.

a. An investigation has been initiated into the possibility of replacing the existing excess flow check valves with more suitable device that would not require valving and will allow for continuous monitoring.

b. The instrument surveillance valving techniques will be reviewed with the instrument technicians prior to startup in order to ensure an understanding of proper valving, and the notification of surpervisory personnel of procedure deficiencies.

c. The performance of surveillance tests will be evaluated on a case by case basis when either startup transformer is out of service.

d. The possibility of procuring an analog instrument system that will not require any valving in order to do surveillance testing is under investigation.

3.D PHYSICAL CHANGES

Platic covers have been placed over the control switches for the suction and discharge motor operated valves. This will require the operator to **lift** the cover prior to actuating the closure of the valve and minimize inadvertent closure of all five suction or discharge valves in the recirculation lines.

4.A Test Program for Startup and Power Ascension

A. General

1. This section provides a brief description of the startup test program. The testing will cover prestartup, startup to hotstandby, and power range testing. Table 4.A-1 is the startup checkoff sheet that will be used during the testing.

B. Prior to Startup

1. Prior to reactor startup, an interference check will be performed on all control rods. This check will ensure that all rods will stroke fully using "Normal" drive pressure, and ensure that all rods will drive in on a scram signal. This testing will ensure that no gross core distortion exists.

C. Normal Startup

1. During the normal startup, the following comparisons and analysis will be performed.

   a. A comparison of the Estimated Critical Position with actual critical configuration ($>173°F$).

   b. An analysis of the reactor coolant after heatup to 250°F.

   c. An analysis of the reactor coolant after heatup to 500 psig.

   d. When primary pressure is approximately 1000psig, a performance check will be made of scram times on Group 1 and Group 2 rods.

2. After reactor startup, with reactor power level raised to approximately 20% rated power, analysis will be performed on the reactor coolant and off gas.

3. As reactor power is increased from 20% to >98% the following analysis will be performed:

	Coolant Iodine	Offgas
30%	X	
40%	X	X
50%	X	
60%	X	X
70%	X	
80%	X	X
90%	X	
>98%	X	X

D. Criteria During Startup to Full Power

1. 0 to 50% of rated power level

a. If the reactor coolant I-134 or I-135 concentrations or stack release rate exceeds those measured at full power prior to shutdown, the reactor will be held at the existing level at which samples were taken until subsequent samples show the criteria to be met. If the sample results exceed a 2 time* increase from the levels measured at full power prior to shutdown, the reactor will be shutdown and placed in the isolated condition until the problem is resolved.

2. 50% to >98% of rated power level

a. If reactor coolant I-134 & I-135 concentrations or stack release rate exceed 1.5 times those measured at full power prior to shutdown, the reactor will be held at the existing level at which samples were taken until subsequent samples show the criteria to be met. If the sample results exceed a 2 time* increase from the levels measured at full power prior to shutdown, the reactor will be shutdown and placed in the isolated condition until the problem is resolved.

E. Reactor Startup Program Checkoff Sheet

1. During startup to full power, each item on the Reactor Startup Program Checkoff Sheet, shown on Table 4.A-1, will be signed off by a cognizant supervisor and a PORC member as it is completed.

F. Full Power Operation

1. The following stack release rate specifications will apply for 14 days after achieving full power operation.

a. Stack Release Rate <1.25 times that prior to shutdown
Follow Technical Specification frequency of sampling analysis.*

b. Stack Release Rate 1.25 to <1.5 times that prior to shutdown
Augmented sampling and analysis program to detect any continuing failures.

c. Stack Release Rate >1.5 times that prior to shutdown
Reduce power level to maintain <1.5.

d. Stackgas Release Rate > 2.0 times
Isolate the reactor and resolve.

* A second sample shall be analyzed to confirm the high values.

* The normal frequency of sampling and analysis of off gas to satisfy Technical Specification requirements is as follows:

1. Samples are taken on Monday, Wednesday, and Friday. They are analyzed for gross gamma activity after 2 hours and 24 hours decay to determine a ratio of long lived to short lived activity.

2. The sample taken on Wednesday is analyzed for noble gas isotopic concentrations.

3. If the ratio obtained in 1 differs from the previous analysis by more than 20%, a new isotopic analysis will be performed.

At full power operation the normal daily reactor water iodine analysis will be performed.

PRE-STARTUP	Date / Time	Supv.	PORC
Interference checks on all control rods			
Reactor coolant analysis - total Iodine, Cl^-, conductivity			

NORMAL STARTUP	Date / Time	Supv.	PORC
Compare ECP with actual critical configuration			
Heat up to 250°F. and analyze coolant iodine			
Heat up to 500# and analyze coolant iodine			
1000 psig - scram time test Group 1 and Group 2 rods			
20% rated power - analyze coolant and off gas			
30% rated power - analyze coolant iodine			
40% rated power - analyze coolant iodine			
40% rated power - analyze off gas			
50% rated power - analyze coolant iodine			
60% rated power - analyze coolant iodine			
60% rated power - analyze off gas			
70% rated power - analyze coolant iodine			
80% rated power - analyze coolant iodine			
80% rated power - analyze off gas			
90% rated power - analyze coolant iodine			
>98% rated power - analyze coolant iodine			
>98% rated power - analyze off gas			

REMARKS:

4.B SUPPLEMENT TO BULLETIN 79-08

Please refer to the response to question 4 to IE-79-08 which was submitted by JCP&L and add the following supplemental information:

The reactor vessel instrumentation described in the above response, except for the low-low-low level instrumentation measures the level in the vessel annulus external to the core shroud. The liquid phase of the annulus reactor water communicates with the core area under the shroud via the reactor recirculation lines while the steam phase in the annulus connects to the shroud through the steam separators which connect to the top of the shroud thus venting the core area to the annulus steam area above the annulus liquid phase. With reactor recirculation pumps stopped and the recirculation line valves open, the mixture level (mass equivalent) within the shroud over the core level is accurately indicated because liquid equalization occurs through the recirculation piping. Additionally, the General Electric response (Attachment 1), verifies that the recirculation flow rate with only one loop open is sufficient to prevent boiloff from reducing water level within the shroud and the reactor will function under normal natural recirculation flow conditions.

4.D ISOLATION CONDENSER OPERATION

The isolation condenser line break sensing system is intended to initiate and achieve an isolation of the condenser within one minute after receiving a line flow signal of 300% or greater. This flow corresponds to a differential pressure (dp) of 27.5 inches of water across the condensate return piping flow sensors and 20 psid across the steam supply piping flow sensors. It is known that transient sensed, high condensate flow rate conditions exist upon isolation condenser initiation. This is due to a surge of cold water stored in the condenser condensate return piping and tube bundle. Since density correction is not performed on the dp signal, the cold higher density water further amplifies the sensed dp. A time delay is incorporated in the system isolation logic to allow for transient dp's in excess of the isolation setpoint for 35 seconds. Based upon this time delay and a valve closure time of 20 seconds, full isolation would occur within one minute of a true line break event.

The isolation condensers return condensate to the suction of A and E recirculation loops. The effects of recirculation flow cause the transient dp to be larger in magnitude and longer in duration. The higher the flow rate, the greater the effect.

Testing performed in November 1972 (report attached) indicates that the normal operating dp across the condensate line flow sensors to be 6-10 inches of water. However, the initial transient dp can exceed 60 inches of water and remain above the isolation setpoint of 27.5 inches of water for 30 seconds from the time of initiation. The testing performed in 1972 was with 5 recirculation pumps in service at minimum flow (~ 4.8×10^4 gpm). It should be noted that subsequent initiations do not yield similar results since the condensate stored in the condenser is hot and the density of the water is less.

4.C EVENT RECORDER OPERATION DURING THE MAY 2, 1979, SCRAM

Description of Event Recorder Operation

Any of 60 signals switch the event recorder from slow speed operation (3/4 inch/hour) into high speed operation (6 inches/minute). Simultaneously, two (redundant) three minute timers are initiated. The event recorder remains in high speed until both of the timers have timed out, at which time slow speed operation resumes. This sequence is independent of signal status.

This sequence of events may be modified by operator action as follows:

a) Moving the on/off switch from on to off to on resets the initiation logic. If all 60 signals have reset, the event recorder will return to slow speed operation regardless of the time after initiation. If any of the 60 signals have not reset, the three minute timers will reset and time for an additional three minute period.

b) Opening the front door of the recorder stops the forward movement of the chart, regardless of the chart speed.

The logic and drive power to the event recorder is from continuous instrument Panel #3. The power to the recording styluses is from the 125 volt DC system.

Recorder Operation on May 2, 1979

All aspects of the event recorder operation were normal following initiation on May 2, 1979. The initiation of the triple low level sensors at 172 seconds is approximately the same time as the completion of the three minute timing sequence. It is evident that the on/off switch was placed in the off position by persons unknown. The event recorder was discovered to be in the off position at approximately 2000 hours of the same day.

A review of available information pertaining to reactor scrams since 1972 demonstrates that when the recirculation pumps are tripped and the isolation condensers are placed in service, inadvertant isolation does not occur. This information is presented in tabular form as follows:

Scram No./Date	Rx Thermal Power (MWt)	Recirculation Flow in GPM (Rated = 16 x 10^4 gpm)	Isolation Condensers Placed in Service	Recirculation Pumps Tripped Yes	Recirculation Pumps Tripped No	Isolation Condenser Isolate Yes	Isolation Condenser Isolate No
#53 April 13, 1972	1857	15.9 x 10^4 gpm	A and B were initiated.	✓ (Lo Lo Level) "C" failed to trip.			✓
#72 September 25, 1974	1920	15.8 x 10^4 gpm	A and B initiated.		✓	✓	
#78 May 4, 1976	1765	14.9 x 10^4 gpm	A and B initiated.	✓ (Lo Lo Level)			✓
#83 September 9, 1978	1310	16.0 x 10^4 gpm	A and B initiated.		✓	✓	
#86 December 13, 1978	1917	16.0 x 10^4 gpm	A and B initiated.	✓ Hi Press			✓
#89 February 2, 1979	1920	15.4 x 10^4 gpm	A and B initiated.	✓ (Lo Lo Level)			✓
#90 May 2, 1979	1895	14.8 x 10^4	B initiated.	✓ (Hi Press)			✓

In reviewing the past scram data, it was never found that when the recirculation system pumps tripped that the isolation condensers isolated. As presented in the tabulation above, five times since 1972, the isolation condensers were initiated without isolation when accompanied by a trip of the operating recirculation pumps.

It is therefore concluded that the isolation condensers will function as intended when accompanied by a trip of the operating recirculation loops. It is noted that the isolation condenser automatic initiation signals are the same as those for automatic recirculation pump trip.

TEST R...

ISOLATION CONDENSER TEST

Date: November 14, 1972

Purpose:

1. Determine the magnitude of the steam and condensate leg d/p spikes
2. Determine the duration of the above d/p spikes
3. Calculate condenser heat removal capacity, if possible

Initial Conditions:

The reactor was in the "RUN MODE" at a power level of approximately 400 MWt. Reactor level control was in the automatic mode and reactor pressure was maintained at 1010 psig by use of the bypass valves under the control of the EPR. Recirculation flow was at the minimum value (20 Hz). The APRMs were set such that 100% was equivalent to 1200 MWt.

Results:

Each isolation condenser was put into service twice and the resulting transients were recorded. Figures 1-A and 1-B are the d/p vs. time plots for the condensers. Each figure contains the d/p's for the steam leg elbow and the condensate return elbow for both Run #1 and Run #2. The pertinent test results are found in Table 1 and comments follow:

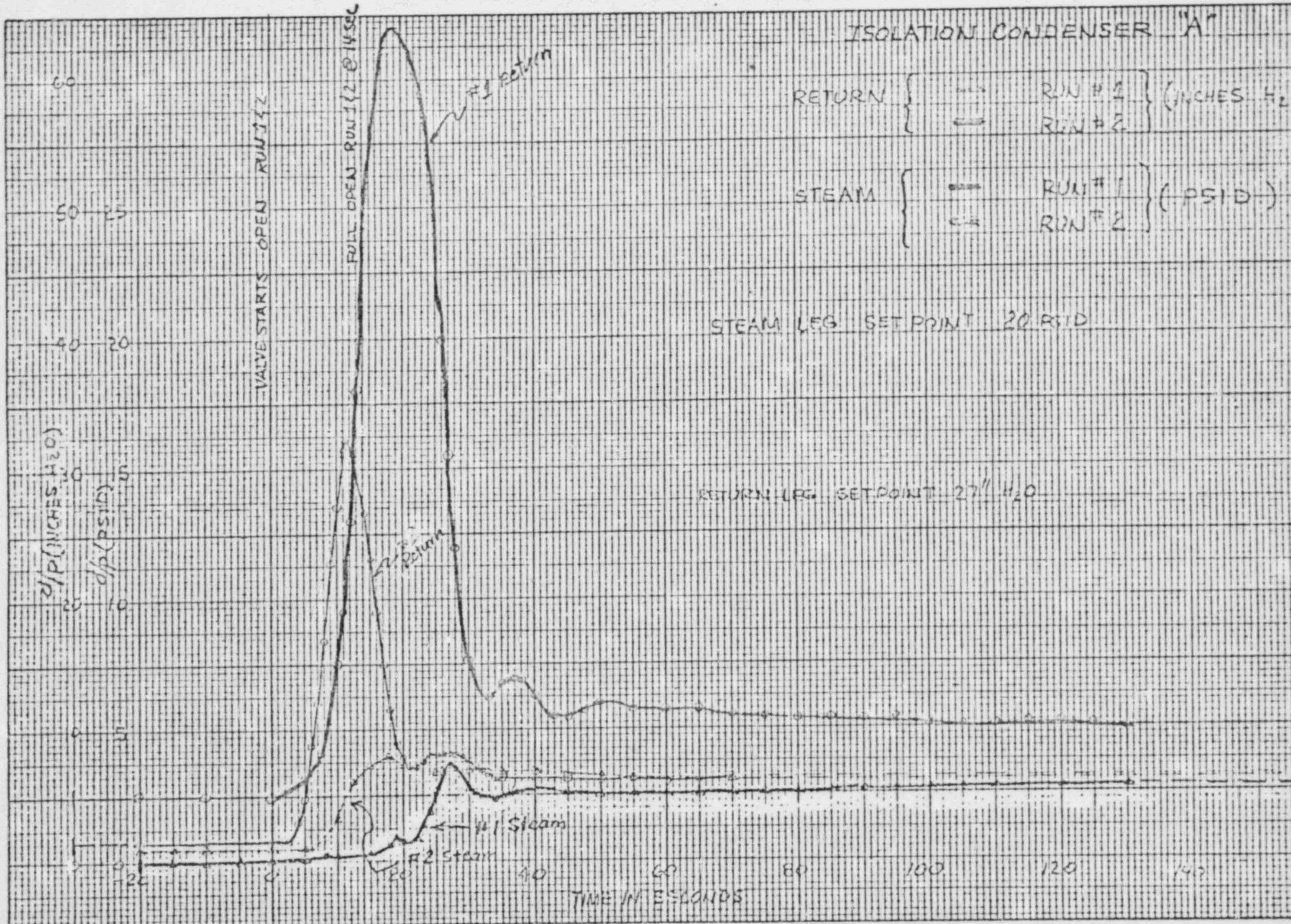
For Run #1, the condensate return leg d/p's for both condensers exceeded the range of the recorder (0-40 inches of H_2O) and also exceeded the range of the local d/p gages (0-60 inches of water). This magnitude for the spike is normal when an isolation condenser has been in the standby condition for a period of time prior to its being put into service. For this condition, the condensate return leg water temperature is cool and the driving head is relatively large even though the condensate return isolation valve may leak slightly. With this valve not leaking, the d/p spikes would probably reach greater magnitudes than observed.

The steam leg d/p for Isolation Condenser "B" was not recorded due to a loose electrical connection at the recorder, which accounts for the missing plot on Figure 1-B. Each condenser was in operation for about 6 1/2 minutes for this run.

For Run #2 all recorded parameters remained on scale. Since Run #2 was performed a relatively short time after Run #1, the condensate return legs had not cooled completely to their normal "in-standby" temperatures. (There is no temperature indication on the return legs) hence, the lower magnitude d/p spikes.

An attempt was made to determine the heat capacity of the Isolation Condensers. The method available to perform the calculation relies on the APRM System to determine the power level when the Isolation Condenser is in operation. At the low power level and non-normal core power distribution experienced during the tests, the APRM System could not be relied upon to provide enough accuracy to make a good determination. Results of the attempts gave indication that the removal capacities were anywhere from 2.7% of rated power to 5.9% of rated power. The condensers are designed for a heat removal capacity of approximately 3% of rated power.

It is recommended that this portion of the test be performed at higher power levels (75%) so that the APRMs and other plant parameters will provide more reliable information. If this is done, however, the heat removal capacity under designed conditions will still not be determined since the recirculation pumps (not running for the design conditions) will be at near rated capacity.

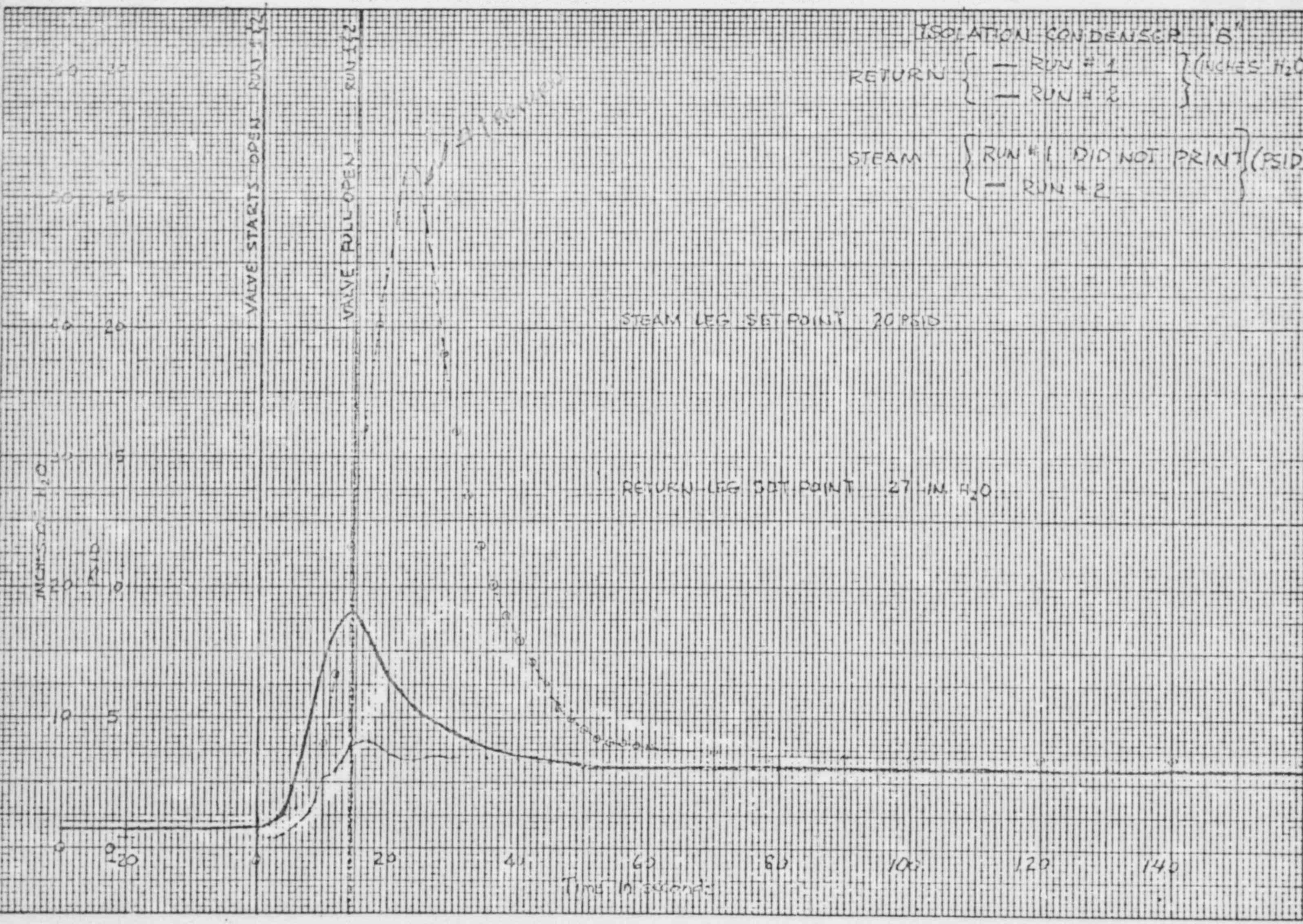

ISOLATION CONDENSER "A"
RETURN { RUN # 1 / RUN # 2 } (INCHES H_2O)
STEAM { RUN # 1 / RUN # 2 } (PSID)
STEAM LEG SET POINT 20 PSID
RETURN LEG SET POINT 27" H_2O
VALVE STARTS OPEN RUN 1&2
FULL OPEN RUN 1&2 @ 14 SEC
#1 return
#2 Return
#1 Steam
#2 Steam
d/p (INCHES H_2O)
d/p (PSID)
60
50 25
40 20
30 15
20 10
5
-20
20
40
60
80
100
120
140
TIME IN SECONDS

JERSEY CENTRAL POWER & LIGHT COMPANY
OYSTER CREEK NUCLEAR GENERATING STATION

DATE: 11/14/72

ISOLATION CONDENSER TEST

		UNIT A Run 1	UNIT A Run 2		UNIT B Run 1	UNIT B RUN 2		
est Duration		7.0 min.	5.2 min.		6.5 min.	5.5 min.		
alve Opening Time		14 sec.	14 sec.		14.5 sec.	14.5 sec.		
nitial Shell Temp.		216 °F	214 °F		216 °F	190 °F		
inal Shell Temp.		236 °F	233 °F		230 °F	220 °F		
t to High Flow Setpoint								
1 Return Leg		12.0 sec.	11.5 sec		14.8 sec.	*		
2 Steam Leg		*	*		*	*		
uration Above Setpoint								
1 Return Leg		15.5 sec.	4.1 sec.		17.0 sec.	*		
2 Steam Leg		*	*		*	*		
eak Δ Pressure								
1 Return Leg		>60 in. H_2O	32.2 in. H_2O		>50 in. H_2O	18 in. H_2O		
2 Steam Leg		3.75 psid	4.0 psid		NOT RECORDED	4.1 psid.		
perating Δ Pressure								
1 Return Leg		10 in. H_2O	6.25 in. H_2O		6.1 in. H_2O	6.25 in. H_2O		
2 Steam Leg		2.75 psid	3.0 psid		NOT RECORDED	3.0 psid		
* DID NOT REACH SETPOINT								



SAME AS STRIP CHART

4.E LEVEL INSTRUMENTATION

Table 4.E-1 (attached) details the indication and recording capabilities as well as auto functions for all reactor vessel level instrumentation installed at Oyster Creek.

In Addition, the GE/MAC narrow range level signals A and B input to plant performance computer.

The narrow range GE/MAC indicators (2) and recorder are the only level monitors that have unique units of level. Their range is zero (0) to eight (8) feet which matches quantatively with the span of "TOTAL FEEDWATER FLOW" signal (zero to 8(x 10^6) lb/hr) which shares a common recorder chart and scale. Consideration had been given to change narrow range level units to inches of H_2O to coincide with units of all other level instrumentation, and to that extent, parts to modify indicator, recorder, and feedwater controller level setpoint scale had been ordered.

TABLE 4.E-1 - SUMMARY OF REACTOR VESSEL WATER LEVEL INSTRUMENTS

SENSOR			INDICATION		ANALOG RECORDER	DIGITAL EVENT RECORDER	ACTUATION OR CONTROL FUNCTIONS	DENSITY COMPENSATION
QTY.	DESCRIPTION	ID NO.	LOCAL	REMOTE				
4	"Low-Low" Level Indicating Switches (Yarway)	RE-02A	Yes	No	No	Yes	1 - Core Spray Init. 2 - Cont. Spray Init. 3 - Reactor Isolation 4 - Containment Isolation 5 - Recirc Pump Trip 6 - Isol. Cond. Init. 7 - SGTS Init. 8 - Annunciators	Compensated during calibration for conditions of operating temperature and pressure.
		RE-02B	Yes			Yes		
		RE-02C	Yes			Yes		
		RE-02D	Yes			Yes		
4	"Low" Level Indicating Switches (Yarway)	RE-05A	Yes	No	No	Yes*	1 - Low Level Scram* 2 - High Level Turbine Trip & Target Relay 3 - Annunciators (Low Level)	
		RE-05/19A	Yes	Yes		Yes*		
		RE-05B	Yes	No		Yes*		
		RE-05/19B	Yes	Yes		Yes*		
4	"Low-Low-Low" Level Indicating Switches (Barton)	RE-18A	Yes	No	No	Yes	1 - Auto Depressurization System Initiation 2 - Annunciators	Setpoint is compensated for weight of steam above variable leg and temperature of reference log.
		RE-18B	Yes			Yes		
		RE-18C	Yes			Yes		
		RE-18D	Yes			Yes		
2	Narrow Range Level (GE/MAC)	ID-13A	No	Yes		No	1 - Feedwater Control 2 - Annunciator (Hi/Low Level) 3 - Feed Pump Runout Protection Reset	Auto density comp. throughout range of level and pressure.
		ID-13B		Yes				
1	Wide Range (GE/MAC)	ID-12	No	Yes		No	None	

NOTES (1): Variable legs of sensors listed above sense level in downcomer region (annulus) except the triple low sensors which sense above core region at core spray sparger.

(2): Reference leg condensate pots tap off of upper downcomer region for all sensors above except wide range level which taps into top of upper head.

4.F FUTURE ACTIONS UNDER CONSIDERATION

As a result of this event and our review of the factors which contributed to the plant trip and subsequent operator action, the following modifications are being considered:

1. Develop sequence of events recording capability which would provide event recording for a period of hours. This might be done by use of a panel alarm automatic logging recorder along with a plant computer capable of monitoring more plant parameters.
2. Evaluate a wiring modification that would require the C.R. operator to hold the control switch for both recirculation pump suction and discharge valves for the two minutes required to fully shut the valve; or electrically interlock the recirculation pump suction and discharge valves to prevent closure of all valves simultaneously.
3. Review what can be done to minimize instrument surveillance testing which might cause a reactor scram while a startup bank is out of service.
4. Continue the engineering review of the desirability of installing a modification that provides solid state sensing devices to replace existing mechanical devices for trip actuation.

5. Investigate procedural and/or excess flow check valve modifications which would eliminate the delicate evolution presently required to insure the check valves are open following a surveillance test.

6. Change the scale readouts on the two existing level inst. readouts in the control room (Yarways and GE/MAC recorder) so they are equivalent, and indicate water level above the core.

7. Investigate ways to improve the reliability of the feedwater system.

8. Provide control room indication of low-low-low water level.

9. Provide overload bypass switches for the feedwater pumps and oil pumps in the control room.

APPENDIX 1

NATURAL CIRCULATION FLOW

A. CALCULATIONS OF INITIAL CONDITIONS

Following scram and pump trips, natural circulation is established. The natural circulation core flow rate is of the order of 10^7 lb/hr and unevaporated water spills over into the downcomer. As the discharge valves are closed, this flow will decay to approximately 200,000-230,000 lb/hr. In addition, control rod drive cooling water is available. The minimum flow at which drainage of water starts occurring in the separators corresponds to the situation where the inlet flow to the core cannot make up the boil-off of steam.

At 3 minutes into the transient,

Core power	=	1895 MW * 0.0333	(May-Witt)
	=	63.1 MW	
Evaporation rate	=	326,000 lb/hr	
Flashing in core	=	38,000 lb/hr	
Total Wg	=	364,000 lb/hr	

Leakage flow from bypass region (1 psid)

= 840,000 lb/hr (thru channel-tieplate leakage path)

Core inlet flow = 840,000 + 260,000 = 1,100,000 lbs/hr

The variation of the flow rates and void fractions vs. elevation are sketched below. The vapor flow rate increases in the core corresponding to power input vs height. The inlet flow is boosted by the leakage from the bypass. At the top of the core the leakage flow is returned to the bypass. This forms a natural circulation loop between the core and the bypass. The net liquid flow in the upper plenum and separators is thus the difference between the core exit flow and the leakage flow.

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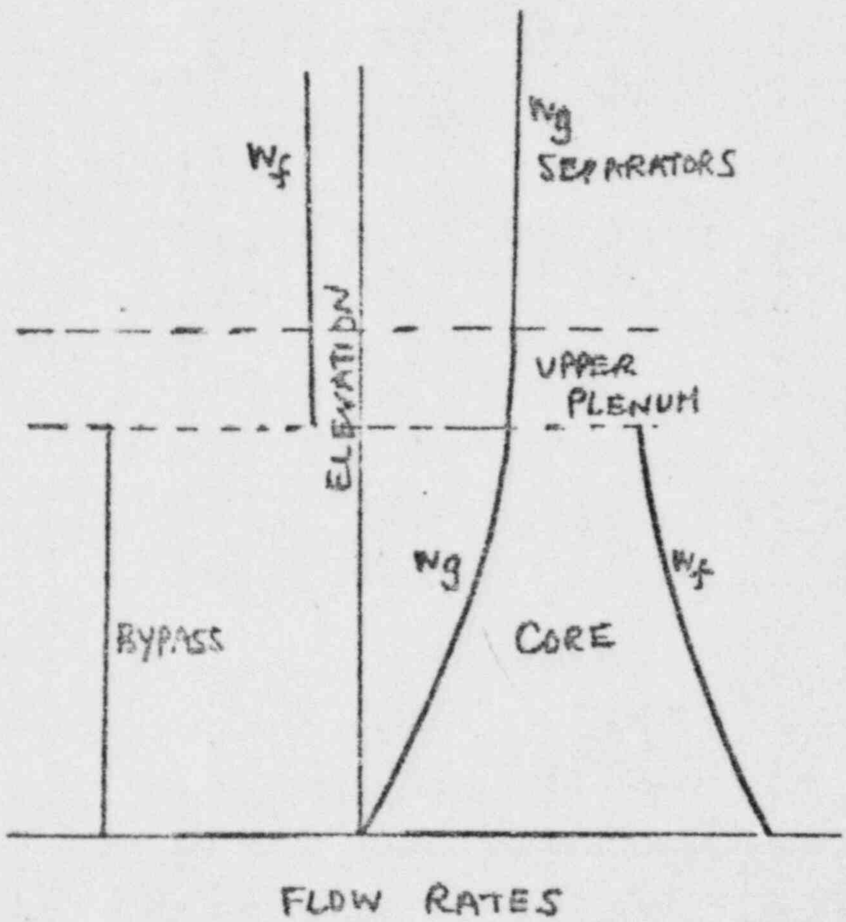
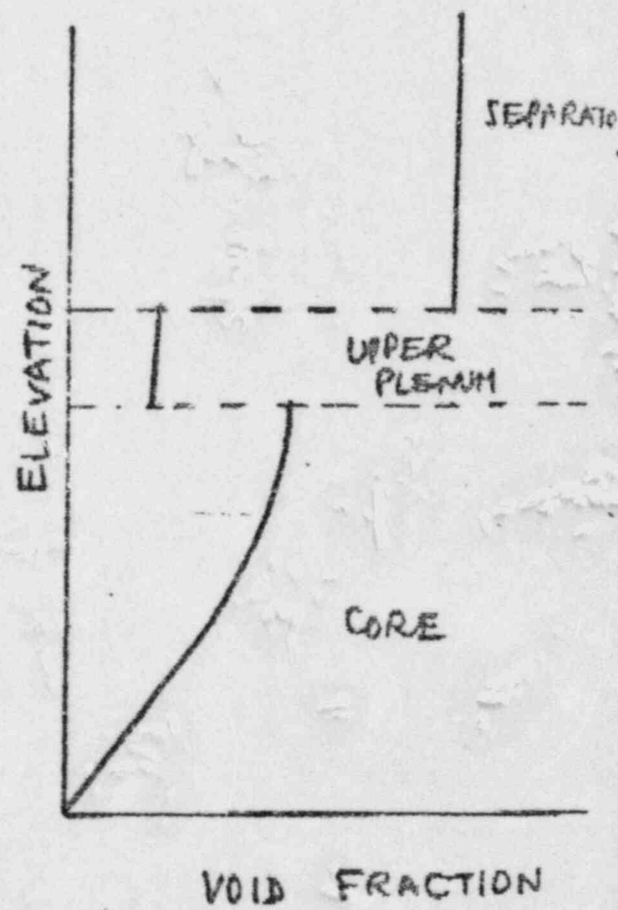
The void fraction increases with height in the core. The large area and low vapor velocity in the upper plenum lead to a lower void fraction in the upper plenum. The vapor accelerates in the separator standpipes producing a high void fraction. The void fraction in the three regions are calculated below:





	CORE (exit)	UPPER PLENUM	SEPARATORS
W_g (lb/hr) (includes flashing)	364,000	443,360	452,000
W_f lb/hr)	736,000	-104,000	-114,000
Drift Flux Parameters C_o	1.3	1.0	1.3
$\bar{V}_{gj}$	0.82 ft/sec	1.0	0.82
Cross sectional area	56 ft^2	186	30.0
$j_g = \frac{W_g}{A\rho_g}$	2900 ft/hr	1123	7100
$j_f = \frac{W_f}{A\rho_g}$	278 ft/hr	-11.8	-8.0
$j = j_f + j_g$	3178	1112	7020
$\alpha = \frac{j_g}{jC_o + \bar{V}_{gj} * 3600}$	0.41	0.24	0.59

Average α for core = 0.21

As the core inlet flow has decreased below the vaporization rate, the level in the separator will fall. The swings in pressure, particularly during pressurization, may accelerate this process.

Before the level falls, the total static heads inside the shroud are:

Above BAF: 13.2 * (1-0.20) + 5*(1-0.25) + 9* (1-0.59) = 14.31 ft.

Above TAF: 1.2 (1-0.4) + 5* (1-0.25) + 9* (1-0.59) = 7.44 ft.

Water inventory inside shroud before separators drain:

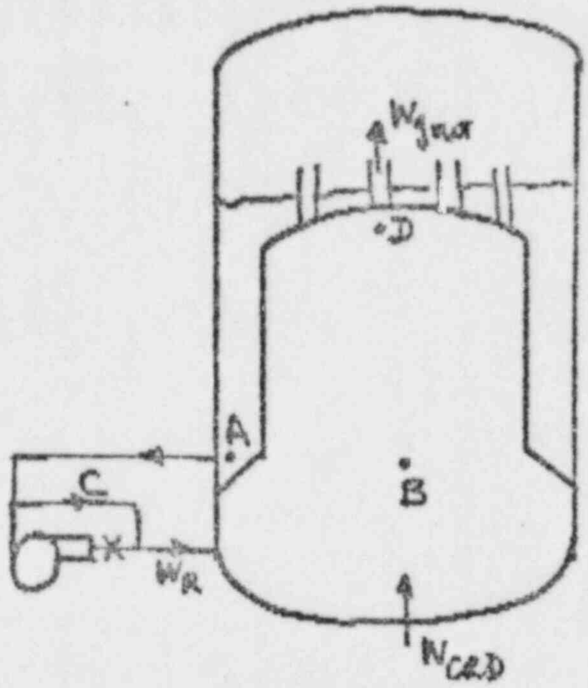
	Volume	Mass (lb)
Bypass:	614	26,955
Core (including unheated region above TAF):	835	28,490
Upper plenum:	783	29,780
Separator standpipes	186	3,561

In order to get the Triple Low Level Alarm, collapsed level must drop to 5.5 ft above TAF. Using the region void fractions this implies that two phase level must be at 2.6 ft into the separator standpipes. This corresponds to approximately 31,600 of fluid above the active core.

Note: BAF - Bottom of Active Fuel
TAF = Top of Active Fuel

B. CALCULATION OF NATURAL CIRCULATION FLOW

The natural circulation flow from the downcomer to the core depends on the static heads inside and outside the shroud:



$$W_{IN} = W_R + W_{CRD}$$

The static head inside the shroud will be history dependent based on the net water flow into the core region. However, based on the results (to follow) that inflow is of the same order as the boiloff, the level is expected to stay near the middle of the upper plenum. The void fraction will gradually decrease inside the shroud as the power falls. However, the two values of recirculation flow chosen should bound this effect.

Pressure drop in the external loop ACB is controlled by the resistance in the five 2" bypass lines. This pressure drop must be balanced by the pressure gain along BDA.

$$\Delta P_{ACB} = \left(\frac{W_r}{5\rho_f A}\right)^2 \frac{K_{bypass}}{2g} \quad \text{(feet of water)}$$

where A = flow area of 2" line (schedule 80) (assumed)
= 2.953 in^2

$$\left.\begin{aligned} K_{bypass} &= 120\ f_t \text{ (for 4 elbows)} \\ &+ \ \ 8\ f_t \text{ (for gate valve)} \\ &+ \ 72\ f_t \text{ (pipe friction)} \\ f_t &= 0.019 \\ \text{Exit loss} &= 1 \\ K &= 4.8 \end{aligned}\right\} \text{Crane Handbook}$$

Substituting these values,

$$\underset{(ft)}{\Delta P_{ACB}} = \frac{W_r^2\ (lb/hr)^2}{0.407 * 10^{10}} \qquad (1)$$

Inside the vessel,

$$-\Delta P_{BDA} = H_{downcomer} - \sum (1-\alpha)_i\ h_i \qquad (2)$$

i = core, upper plenum, separators.

Assuming the void fractiors calculated earlier and an average level at the midddle of the upper plenum,

$$\Delta P_{BDA} = H_{downcomer} - 13.0 \qquad (5)$$

Level in the downcomer ranges from 136" to 170" above the top active fuel (i.e., 280" to 310" above BAF).*

Using, $\Delta P_{ACB} = -\Delta P_{BDA}$,

$$W_r^2 = 0.407 * 10^{10} * (H_d - 13.0) \quad (4)$$

For the low downcomer level of 280" (23.3 ft).

$$W_r = 205,000 \text{ lb/hr}$$

For the high downcomer level of 310" (25.85 ft).

$$W_r = 229,000 \text{ lb/hr}$$

The uncertainty in this calculation is primarily the static head inside the shroud. However, since a smaller static head produces a larger recirculation flow, compensating effects are introduced. In any event, the recirculation flow (together with CRD flow) should be sufficient to make up the boiloff after the first few minutes.

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*More accurate interpretation of the level accounting for density changes and flashing in the downcomer below the pressure tap location leads to a variation of 270" to 310" above BAF.

Inventory Inside the Shroud

The inventory calculation has been improved by consideration of subcooling of the lower plenum, flashing and stored energy effects.

Mass balance inside shroud:

$$M = W_{in} - Wg_{out} \qquad (1)$$

W_{in} = Natural recirculation flow + CRD flow

Wg_{out} = Steam outflow due to flashing and evaporation

$$M_t = M_{initial} + \int_{ti}^{t} (W_{in} - Wg_{out})dt \qquad (2)$$

M_t = Total mass inside shroud at time t

$M_{initial}$ = Total mass inside shroud at 3 minutes

The region inside the shroud is composed of four regions:

1. Lower plenum and control rod guide tubes. This region will remain single phase liquid, but change in temperature.
2. Core region
3. Bypass region
4. Upper plenum (and separators).

Lower Plenum and Guide Tubes

$$\frac{dE}{dt} = W_{in}h_{in} - W_{out}h_{out} + Q \qquad (3)$$

E = Total internal energy in lower plenum

h_{out} = enthalpy leaving lower plenum

Q = heat from vessel walls

Various assumptions can be made about h_{out}. If the plenum is perfectly mixed, h_{out} corresponds to the average plenum temperature. The worst case for the level calculation in the upper plenum (i.e. the one that yields the lowest level) is to assume perfectly stratified flow. The outflow enthalpy in this case remains the initial enthalpy until the initial mass in the plenum is entirely replaced.

$$M_{LP} = V_{LP}\rho_{LP} \qquad (4)$$

$$\dot{M} = W_{in} - W_{out} = V\dot{\rho} = Vh\frac{d\rho}{dh} \qquad (5)$$

Approximating internal energy by enthalpy, equation 3 becomes

$$\dot{(Mh)} = W_{in}h_{in} - W_{out}h_{out} + Q \qquad (6)$$

Substituting (5) into (6)

$$\dot{h} = \frac{W_{in}(h_{in} - h_{out}) + Q}{M - V\frac{d\rho}{dh}(h_{out} - h)} \qquad (7)$$

$h_{out} = h_{initial}$ for $\frac{M_{in}}{W_{in}}$ = 27 minutes

Table 3.1 shows the calculation of lower plenum temperature and mass using this procedure. The heat from the vessel wall was estimated to be fairly small and neglected. This is conservative for the level calculation.

Two-Phase Regions

From the energy equation, vapor generation rate in a particular region

$$\Gamma_g = \frac{1}{h_{fg}} \left[Q + V(\frac{1}{J} \rho_g \alpha \frac{dh_g}{dp} - \rho_f (1-\alpha) \frac{dh_f}{dP} \dot{P} \right] \quad (8)$$

In the core both negative and positive $\dot{P}$ are used; in the bypass and upper plenum only depressurization is considered.

The steam flow leaving the core and bypass regions (and generated in the upper plenum) is calculated assuming quasi-static conditions (i.e., $\frac{d\alpha}{dt}$ small).

$$W_g = (\Gamma_g V_g + \alpha \rho_g \frac{dv_g}{dp} \cdot \dot{P}) \rho_g \quad (9)$$

The steam flow leaving the shroud

$$Wg_{out} = Wg_{core} + Wg_{bypass} + Wg_{upper\ plenum} \quad (10)$$

The mass in each region is calculated as below:

$$j_g = \frac{\Gamma_g v_g}{A} + \alpha \rho_g \frac{dv_g}{dp} \cdot \dot{P} L \quad (11)$$

$$\alpha_{exit} = \frac{j_g}{JC_o + V_{gj}} \quad (12)$$

$$\bar{\alpha} = 0.5 \, (\alpha_{exit} + \alpha_{inlet}) \quad (13)$$

$$M = \left[\bar{\alpha} \rho_g + (1-\bar{\alpha}) \rho_f \right] V \quad (14)$$

Table 3.2 lists the system parameters used in the calculations. The pressures and pressure rates were obtained from the data from the site. The decay heat factors correspond to the May-Witt curve. It is estimated that stored energy release from the core is less than 5% of the decay power after the first three minutes. Since the May-Witt values are conservative by 16-23% compared to the new best estimate ANS Stand , decay heat values more than compensate for stored energy effects.

Using the data in Table 3.2, the steam flow rates and void fractions were calculated as a function of time for the various regions, as given by equations 8-14. These are tabulated in Table 3.3.

Tables 3.1 and 3.3 provide sufficient information for a mass balance using Equation 1.

$$\text{Then,}\quad M_{\text{upper plenum}} = M_t - M_{\text{core}} - M_{\text{bypass}} - M_{\text{lower plenum}} \qquad (15)$$

Table 3.4 shows the calculation of total mass and upper plenum mass, using equations (1), (2), and (5). Two values of recirculation flow at 200,000 and 230,000 lbs/hr were used. At the higher flow, a minimum mass of 20,000 lbs (33" collapsed level) was reached at between 9 and 15 minutes. Beyond this time, the inflow is able to overcome the effects of vapor generation and increased density in the lower plenum.

For the lower flow, a minimum value of approximately 16,000 lbs (28" collapsed level) was reached at 11-23 minutes. An increase occurs thereafter.

The variation in mass and collapsed level is shown in Figure 1.

Table 3.1

Lower Plenum Mass Calculation

$$\dot{h} = \frac{4583\ (h_{in} - h_{out}) + Q}{1.521\times10^5} \quad \text{(BTU/lb/min)}$$

(using $\overline{M}$ - 1.43×10^5 lbs, $\frac{dP}{dh}$ = -0.04188 between h = 512 to 356 BTU/lb

h_{out}-h = 77 BTU/lb. h_{in} calculated by weighting enthalpies of recirc and CRD flow)

Time	h_{in}	h_{out}	Q BTU/min	$\dot{h}$	$\bar{h}$	v_L	Mass
3	410	512	NEGLECTED	-3.07	512	0.02091	1.341×10^5
4	410	512		-3.07	509	0.02086	1.345×10^5
5	410	512		-3.07	506	0.0208	1.348×10^5
6	410	512		-3.07	503	0.0207	1.355×10^5
7	410	512		-3.07	500	0.02067	1.357×10^5
8	410	512		-3.07	496.5	0.0206	1.362×10^5
9	410	512		-3.07	493.5	0.0205	1.368×10^5
10	410	512		-3.07	490.5	0.02048	1.369×10^5
12	382	512		-3.92	484.5	0.02039	1.375×10^5
14	362	512		-4.52	476.7	0.02025	1.385×10^5
16	340	512		-5.18	467.6	0.02006	1.398×10^5
18	330	512		-5.48	457.2	0.01988	1.411×10^5
20	320	512		-5.78	446.2	0.01967	1.426×10^5
25	313	512		-6.00	417.3	0.01924	1.458×10^5
30	306	410		-3.28	387.3	0.0188	1.492×10^5
33	301	410			377.5	0.01867	1.502×10^5

TABLE 3.2

Time (min)	P (psig)	$\dot{P}$ (psi/sec)	Qd (pu)	Qd (mW_T) Based on 1895 mW_T	h_{fg} (Btu/lb)	W_B (lb/sec) $W_B = 1000Qd/1.054h_{fg}$
3	920	-1.3	0.0346	65.57	660.9	94.13
4	850	-1.3	0.0326	61.78	674.6	86.89
5	850	1.3	0.0310	58.75	674.6	82.63
6	920	1.3	0.0294	55.71	661.0	79.96
7	1000	1.3	0.0280	53.06	647.5	77.75
8	950	-1.65	0.0272	51.54	657.0	74.43
9	830	1.65	0.0263	49.84	680.5	69.49
10	780	-1	0.0255	48.32	690.6	66.38
12	730	-0.23	0.0244	46.24	700.8	62.60
14	710	-0.23	0.0234	44.34	704.9	59.68
16	680	-0.23	0.0226	42.83	711.1	57.14
18	650	-0.23	0.0220	41.69	717.7	55.11
20	620	-0.23	0.0215	40.74	724.2	53.37
25	760	0	0.0204	38.69	694.6	52.85
30	620	0	0.0192	36.38	724.2	47.66
33.33	510	0	0.0184	34.89	749.3	44.18

TABLE 3.3

Time (min)	Flashing Rate Core (lb/sec)	Bypass (lb/sec)	Upper Plenum (lb/sec)	α_{core}	α_{Bypass}	α_{up}	M_{core} (lb)	M_{By} (lb)	M_{up} (lb)
(3)	10.7	9.9	12	.21	.065	.25	26,600	26,955	31,590
4	10.7	9.9	12	.21	.065	.25	26,600	26,955	
5	- 10.7	0	0	.20	0	.24	26,900	29,570	
6	- 10.7	0	0	.19	0	.23	27,236	29,570	
7	- 10.7	0	0	.19	0	.22	27,236	29,570	
8	10.7	0	12	.20	.065	.23	26,600	28,430	
9	10.7	0	0	.20	0	.21	26,600	29,300	
(10)	6.4	4.7	5.8	.18	.04	.19	28,290	28,430	
12	1.5	.97	1.1	.17	.01	.17	28,620	30,000	
14	1.5	.97	1.1	.17	.01	.16	28,620	30,000	
(16)	1.5	.97	1.1	.16	.01	.15	29,590	30,000	
18	1.5	.97	1.1	.16	.01	.15	29,590	30,000	
20	1.5	.97	1.1	.15	.01	.14	29,913	30,000	
25	0	0	0	.15	0	.13	29,913	30,296	
(30)	0	0	0	.14	0	.13	30,236	30,296	
33.33	0	0	0	.14	0	.13	30,236	30,296	

TABLE 3.4(a)

Oyster Creek Inventory Calculation - 1

Recirc Flow = 230,000 lb/hr

Time (min)	W_{out} (lb/sec)	W_{in} (lb/sec)	$\dot{M}$ (lb/sec)	M (lb)	M_{up} (lb)
3	126.7	79.2	-47.5	219,245	31,590
4	119.5	79.2	-40.3	216,6111	28,256
5	71.9	79.2	7.3	215,620	23,650
6	69.3	79.2	9.9	216,130	23,824
7	67.1	79.2	12.1	216,790	24,284
8	97.1	79.2	-17.9	216,442	25,512
9	80.2	79.2	- 1.0	215,876	21,156
10	83.3	79.2	- 4.1	215,726	22,106
12	66.2	79.2	13.0	216,222	20,140
14	63.3	79.2	15.9	217,956	20,874
16	60.7	79.2	18.5	220,020	20,668
18	58.7	79.2	20.5	222,398	21,708
20	56.9	79.2	22.3	214,968	22,455
25	52.9	79.2	26.3	232,218	26,249
30	47.7	79.2	31.5	240,928	31,194
33.33	44.2	79.2	35.0	247,561	36,830

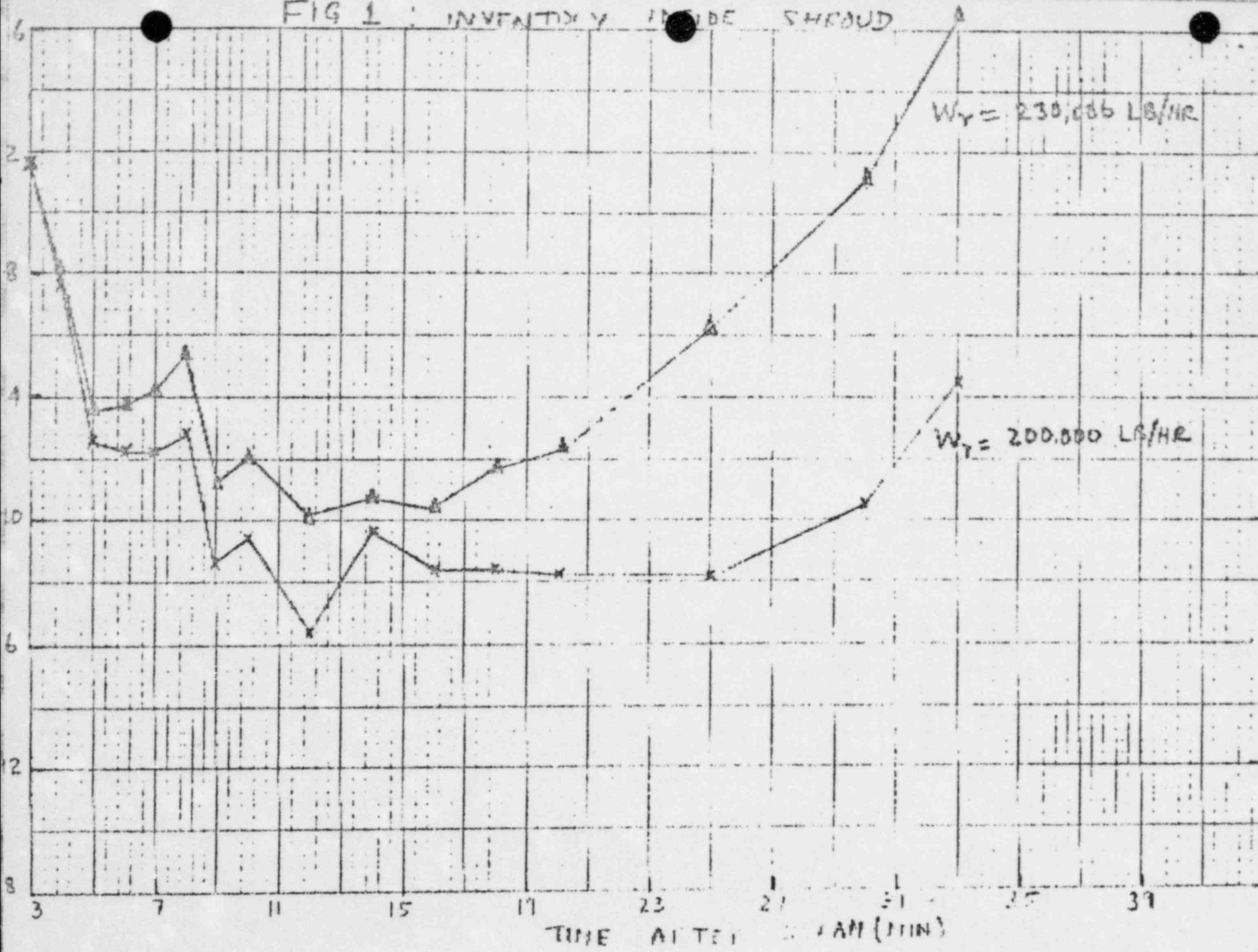
TABLE 3.4(b)

Oyster Creek Inventory Calculation - 2

Recirc Flow - 200,000 lb/hr

Time (min)	W_{out} (lb/sec)	W_{in} (lb/sec)	$\dot{M}$ (lb/sec)	M (lb)	M_{up} (lb)
3	126.7	70.9	-55.8	219,245	31,590
4	119.5	70.9	-48.6	216,113	27,758
5	71.9	70.9	- 1.0	214,613	22,643
6	69.3	70.9	1.6	214,631	22,325
7	67.1	70.9	3.8	214,793	22,287
8	97.1	70.9	-26.2	214,121	22,891
9	69.5	70.9	1.4	213,377	18,657
10	83.3	70.9	-12.4	213,047	19,426
12	66.2	70.9	4.7	212,585	16,465
14	63.3	70.9	7.6	216,792	19,672
16	60.7	70.9	10.2	217,860	18,470
18	58.7	70.9	12.2	219,204	18,514
20	56.9	70.9	14.0	220,776	18,263
25	52.9	70.9	18.0	224,076	18,067
30	47.7	70.9	23.2	230,256	20,524
33.33	44.2	70.9	26.7	235,241	24,508

FIG 1 : INVENTORY INSIDE SHROUD



Oyster Creek

- Addendum

C. Calculation of Natural Circulation Flow with 1 (one) External Loop Open

Utilizing the same methods of sections A + B, the natural circulation flow rate can be calculated by:

$$\Delta P_{ACB} = (W_r/\rho A)^2 \ (K_{loop} + K_{pump}/2g)$$

where:

$\Delta P_{ACB} = -\Delta P_{BDA} = H_{downcomer} - 19.0$ (ft.)

W_r = Flow through open loop, lbs/hr

ρ = density of water in loop, = 47 lbs/ft^3

A = 3.13 ft^2 (26" schedule 80 pipe)

K_{Loop} = 1.25 vessel entrance + exit losses

+1.20 5 elbows

+0.50 2 gate valves

+0.80 1 flow element

+0.25 straight pipe

K_{Loop} = 4.0 total loss coeff. (Crane HdBK values)

$K_{recirc\ pump}$ = 21.0 fixed rotor (worst case vs free rotor) (from Byron-Jackson Test Data)

K_{Total} = 25.0

19.0 ft = static head inside shroud at core flow of $\sim 2 \times 10^6$ lbs/hr

Solving For W_r:

$$W_r = \rho A \ (2\ g/K_{total})^{1/2} \ (H_d - 19.0)^{1/2}$$

$$W_r = 0.85 \times 10^{+6} \ (H_d - 19.0)^{1/2}$$

For the low downcomer level of 280" (23.3')

$$W_{r_L} = 1.76 \times 10^6 \text{ lbs/hr}$$

For the high downcomer level of 310" (25.8')

$$W_{r_H} = 2.22 \times 10^6 \text{ lbs/hr}$$

Since the other four loops would be supplying fluid through the 2" bypass lines, the total flow to the core would be:

$$W_{r_L} = 1.76 \times 10^6 + (0.8)0.205 \times 10^6 \qquad \underline{W_{r_L} = 1.93 \times 10^6 \text{lbs/hr}}$$

$$W_{r_H}\ 2.22 \times 10^6 + (0.8)\ 0.229 \times 10^6 \qquad \underline{W_{r_H} = 2.40 \times 10^6 \text{ lbs/hr}}$$

Thus the recirculation flow rate (which is about 5 to 6 times boiloff rate) with only one loop open is sufficient to prevent boil-off from reducing water level within the shroud and the reactor will function under normal natural circulation flow conditions. With additional loops open, this flow rate would be much greater (approximately 2 x for 2 loops, 3 x for 3 loops, etc) and thus provide even greater margin to boil off.