ACRS SM-0093
PDR 5/22/79

DRAFT 1
4/15/79

REPORT OF TRIP TO THREE MILE ISLAND NUCLEAR STATION

UNIT 2 BY R. F. FRALEY ON APRIL 12, 1979

Persons contacted

NRC	ACRS	Met. Edison
D. Denton	H. Etherington	J. Herbein, V.P
Hoverkamp, Region I	C. Michelson	J. Floyd, Unit 2 Ops. Supervisor
R. H. Vollmer	I. Catton	
B. Grimes	T. G. McCreless	

Industry Support Group	Other Visitors
J. McMillen, B&W	Commissioner Bradford
Thiesing, Bechtel	Commissioner Ahearne
	H. Thompson, ~~NRC~~ OCM
	M. Campbell, OCM
	G Sauter, OCM

Morning

Plant status (9:30 A.M)

Forced circulation is continuing. Late yesterday a degassing operation was run and the primary system pressure was reduced to approximately 300 psi. The pumps and letdown system performed adequately. The pressure was limited by the desire to maintain system pressure at T_{sat} + $100^{o}F$ at the hottest core thermocouple. All core thermocouples are below $400^{o}F$.

When I arrived, work was continuing to get the Heise gage operating as a backup to the pressurizer level indicators. Two of the three ΔP indi-

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are still operating but one has failed and then started operating again. The indicators have already exceeded their allowable not ~~not~~ level and continued operation is of concern.

A sample of primary coolant taken on Wednesday has been analyzed by the Bettis Laboratory and indicates that little, if any, fuel melting occured at the surface in contact with primary coolant although it is possible that some melting may have occurred at the center of the pellets. The analysis of primary coolant is attached. The absence of uranium (<10 ppb) and transuranics provides the most convincing conclusion that "no" core melting occurred.

The loss of the component coolant water to the 1A primary coolant pump was actually an instrument failure rather than a loss of coolant and the pump is again available for sevice. The 2A pump is still providing circulation however.

<u>Meeting with Industry Support Group Members</u>

Contingency plans were still being developed to make the shift from forced to natural circulation. A low pressure lube oil cooler will be connected across the B loop steam generator and a high pressure heat exchanger will be installed across the A loop steam generator. Both steam generators will be flooded to act as heat exchangers rather than steam generators. A series of specific steps are being planned to make the shift from forced to natural circulation to permit the flow to ~~provide for~~ reverse ~~flow~~ in the B loop and to flood the A steam generator without water hammer.

Estimates of core damage indicate that a significant portion of the core has experienced serious damage. An analysis of time-pressure-temperature- etc. curves indicates that the bottom half of the core may have been uncovered for approximately 15 minutes. The upper half may have been uncovered for a total (several intervals) of 60-75 minutes. The initial release of fission products from the fuel is believed to have occurred approximately 73 minutes into the transient (after the primary coolant pumps were secured by the operator).

Visit to Nuclear Plant

The initiator of the transient is now believed to be a loss of control air which permitted valves to close in the condensate system. This caused the condensate pumps to trip and the subsequent loss of main feedwater.

When asked about the existence of any precursors to this transient, the TMI-2 Chief Inspector indicated that on two previous occasions pressurizer level has been "lost" (out of range on the level indicators) due to stuck open relief valves (once on the steam generators and once on the pressurizer). Analysis of the incidents indicates that fluid did exist in the bottom of the pressurizer and/or surge line and the pressurizer bubble did not get out into the primary loop although this was a concern at first.

Operation of the plant and recovery operations appeared to be orderly with the plant operators running the plant and the industry support group providing technical support to evaluate plant status (e.g., core thermocouple readings) and any proposed changes in plant conditions. Metropolitan Edison personnel are making plant modifications such as changing the auxiliary building charcoal filters and installation of a backup bank of charcoal filters on the roof of the Auxiliary Building. Westinghouse personnel are constructing and will install some features such as backup to the residual heat removal system. Preparations were underway to change the installed charcoal filters with filters provided from a WPPSS plant. The concurrence chain for any changes involves the Industry Support Group, Metropolitan Edison, B&W, and NRC but appears to be working smoothly. The implementation of agreed to changes apparently is sometimes delayed, however, because of caution/concern by Metropolitan E [illegible] respect to perturbation of existing operations.

We were shown the three auxiliary feed pumps (2 electric and 1 steam) which were valved out inadvertently during the early phases of the transient (first 8 minutes.) Each pump has a local block valve to isolate it for testing, etc. and two remotely operated (controlled from the reactor console) to block flow to the A and B steam generators. The arrangement is slightly different in the #1 Unit vs. the #2 Unit, but this did not appear to contribute to the accident.

We were shown the control room and specifically the controls and instruments associated with this transient which were available to the operators. The following points are of particular interest.

- The two auxiliary feedwater block valves were closed when the transient started. They were opened manually by the operator when he observed that the auxiliary feed pumps had come on the line but the steam generators were not refilling. One of the valves was tagged out at the time and the tag may have obscured the valve position (open/closed) lights. The other valve was not tagged out. I noted that the valve position switch on both these valves is spring return to the neutral position while others on the panel remain in the open or closed positions providing indication of valve position even if the valve position lights are not visible.

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(There are a mix of these two types of switches on the panel. The Metropolitan Edison representative indicated that he did not feel this contributed to any confusion by the operator regarding the valve position. He noted that the pump block valves are exercised once each week to check the auxiliary pumps operation. This involves 3 weeks and the 4th week a valve position check is made. He was convinced, based on discussion with the personnel who make these checks, that the block valves were open 24 hours prior to the transient.

- The pressurizer relief valve was blocked by the operator at 2.3 hrs into the transient when he realized that he had a primary system leak and attempted to isolate all possible leak paths. Mr Floyd indicated that there is no direct indication available to the operator that the primary relief valves are open. This must be inferred from the R.C. drain tank level indication which is located around behind the instrument cabinets some distance (est. 30 ft.) from the control console.

- The core outlet thermocouples do not read out directly but are printed out by the computer printer on a cycle of approximately

20 minutes during normal operations. During abnormal conditions (e.g., transients) the printer may be as much as 1 hour behind because of the large number of off normal conditions it must scan and the limited speed of the printer. The operator can call for the thermocoule temperatures at any time if he desires. Discussion with the Metropolitan Edison representatives indicated that they did not consider this an important operating parameter.

Plant status (4:00 P.M.)

Plant status had not changed significantly from its status in the morning except that the Heise gage was now operational and a single level check compared to the ΔP indicators was within 10 inches (135" level vs. 145").

Attachments:

1. Preliminary Primary Coolant Analysis for 4/11/79 Sample
2. Preliminary - Description of Events at the TMI2 Facility Accident dtd. 4/10/79
3. TMI Tour and Briefing dtd. 1/78

Vollmer

Preliminary Primary Coolant Analysis $2.2^2 \times 10^6$ µCi

for April 11, 1979 Sample

	Results from Oak Ridge		Results from SRL
Boron	>500 ppm		2600 ± 200 mg/l
pH	8.0		7.0
	Calc. 8:AM		

Radionuclide	µCi/ml	$\times 10^8$	dpm/ml
Mo^{99}	179	4	2.77×10^8
I^{131}	8.2×10^3	1.8	1.01×10^{10}
I^{132}	<20		--
Cs^{134}	82	1.8	1.68×10^8
Cs^{137}	330	7.3	7.11×10^8
Cs^{136}	108	2.4	2.74×10^8
Ba^{136}	--		2.01×10^8
Ba^{140}	290	6.4	3.8×10^8
La^{140}	160	3.6	3.0×10^8
Sr^{89}	600	1.4×10^9	3.3×10^9 dpm/ml
Sr^{90}	50		
Ru^{103}	n.d.		n.d.
Ce^{144}	--		2.28×10^8
H^3	1.2		-
gross β	1.4×10^{10} dpm		9×10^9 cpm/ml
U	<10 ppb		<.001 mg/l
α	n.d.		$<2 \times 10^3$ dpm/ml
Pu	n.d.		no transuranics

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PRELIMINARY April 10, 1979

DESCRIPTION OF EVENTS
AT THE THREE MILE ISLAND 2
FACILITY ACCIDENT

The following is a summary of the significant events that occurred at the Three Mile Island No. 2 nuclear facility on March 28, 1979, and thereafter. Attached is a detailed chronology of these events listed with the times they each occurred.

At about 4:00 am on March 28, 1979, the secondary (nonnuclear) cooling system of the Three Mile Island facility suffered a malfunction. This system normally pumps water through the plant's steam generators where the water turns to steam which then flows to turn a turbine generator. The water is then condensed back to water, is pumped by a condensate pump through a clean up system, through a feedwater pump, and finally back to the steam generators, and continually flows around this loop.

A malfunction in the main feedwater system caused the feedwater pumps to turn off (trip), which in turn caused the turbine-generator to turn off and stop generating electricity. Since the steam generators were not removing heat due to the stoppage of feedwater flow, the reactor coolant system pressure increased and the pressurizer relief valve opened to reduce reactor pressure. Immediately, the reactor turned off by the rapid insertion of the plant's control rods (scrammed) as designed and the nuclear chain reaction stopped leaving behind only residual, or decay, heat. These events all occurred within the first 30 seconds of the accident.

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Up to this point, this sequence is normal and the auxiliary feedwater system should startup and deliver secondary coolant to the plant's two steam generators to remove heat. In addition, the pressurizer relief valve should close as reactor pressure decreases.

All three of the auxiliary feedwater pumps started but were unable to deliver flow because their flow paths were blocked by closed valves. In addition, the pressurizer relief valve failed to close and therefore allowed the reactor coolant system pressure to continue to decrease.

As the reactor pressure reached a preset value (1600 psi), the plant's Emergency Core Cooling System (ECCS) started as designed and began to inject cold water into the reactor about 2 minutes after the event started. An indication of a rapidly rising pressurizer level apparently led the plant operators to terminate the ECCS flow. At this point the Three Mile Island accident had been underway for 10-11 minutes.

Between about 1 and 2 hours into the accident, the operators turned off the four large pumps which circulate the reactor coolant through the reactor. It is following this action that we believe the severe damage to the nuclear fuel began. For the next several hours there was a very large temperature difference across the nuclear core indicating little flow of coolant through the core.

During this several hour period, when severe fuel damage was occurring, primary coolant from the reactor primary coolant system was being dumped onto the reactor containment floor from flow out of the pressurizer relief valve and through the drain tank. This coolant, which contained radioactivity, was partially pumped from the reactor containment building floor to tanks in the auxiliary building. The tanks overflowed permitting radioactivity to be vented from the auxiliary building. This situation lasted until about 9:00 am when the reactor containment was sealed (isolated).

From about 6:00 a.m. until 8:00 p.m., the licensee tried to depressurize the reactor coolant system sufficiently to be able to turn on the residual heat removal system. Since his attempts failed, it was decided to repressurize the system.

After repressurization, one of the main reactor coolant pumps was restarted and flow through the reactor core was re-established.

Since feedwater was being provided to the steam generator, heat was being removed and the reactor system was slowly cooled. Core temperatures decreased over the next several days and stabilized. Reactor cooling has essentially been in this mode since that time.

PRELIMINARY CHRONOLOGY OF THE MARCH 28, 1979 ACCIDENT AT THREE MILE ISLAND

Time (approximate)	Discussion of Events
Before 4:00 a.m.	TMI operator working on Feedwater System
4:00 a.m.	The loss of all (main and auxiliary) feedwater flow occurred while tha reactor was operating at 98% power. The transient was initiated by a loss of condensate pumps. The turbine tripped.
3-6 sec later	An electromatic relief valve opened to relieve pressure in the RCS* (2255 psi).
9 sec after start of event	The Reactor tripped on high RCS pressure (2355 psi) to terminate the nuclear reactor and reduce power generation to decay heat alone.
12-15 sec after start of event	The RCS pressure decayed to the point (2205 psi) where the relief valve should have reclosed. The RCS continued to depressurize for about the next two hours.
14 sec after start of event	The auxiliary feedwater pumps in both safety trains (1 turbine driven pump and 2 electrically driven pumps) were started and were running at pressure ready to inject water into the steam generators and remove the residual heat produced in the reactor core. No water was injected since the discharge valves were closed.
15 sec after start of event	The temperature in the RCS hot leg peaks at about 610°F with a pressure of about 2150 psi.

*Throughout, RCS denotes "reactor coolant system."

Time (approximate)	Discussion of Events
4:01 a.m.	The pressurizer level indication began to rise rapidly. The steam generators, A and B, had low levels of water and were drying out.
4:02 a.m.	The ECCS was initiated as the RCS pressure decreased to 1600 psi.
4:06 a.m.	The pressurizer level indication went offscale high.
4:04-4:11 a.m.	The operator manually tripped the first HPI pumps at about 4:05:15 and the second at about 4:11:01.
4:06 a.m.	Water in the RCS flashed to steam as the pressure bottoms out at 1350 psi. The hog leg temperature was about 584°F.
4:07-4:08 a.m.	The Reactor building sump pump came on.
4:08 a.m.	The operator opened the valves at the discharge of the auxiliary feedwater pump allowing water to be injected into the steam generators.
4:12-4:13 a.m.	The operator restarted the ECCS to inject water into the RCS to control pressurizer level.
4:11 a.m.	The pressurizer level indication comes back on scale.
4:15 a.m.	The RC Drain (Quench) tank rupture disk blew at 190 psig due to continued discharge of the relief valve that had failed to close.
4:20-5:00 a.m.	The RCS parameters stabilized at a saturated condition of about 1015 psi and 550°F.
5:14 a.m.	The operator tripped both RC pumps in Loop B and one pump in Loop A.
5:27 a.m.	Operator isolated "B" Steam generator.
5:41 a.m.	The operator tripped the second RC pump in Loop A.

Time (approximate)	Discussion of Events
5:45-6 a.m.	The reactor core began a heatup transient. The RCS hot leg temperature went offscale at 620 degrees F within 14 minutes and the cold leg temperature dropped to near the temperature of high pressure injection water (150 degrees F).
6:20 a.m.	The failed c[illegible] relief valve was isolated by the oper[illegible] by closing a block valve.
7:00 a.m.	The RCS pressure had increased to 2150 psi and the relief valve was opened to relief RCS pressure.
7:15 a.m.	A pressure spike of 5 psig occurred in the RC drain tank due to steam from the relief valve.
7:45 a.m.	A pressure spike of 11 psig occurred in the RC drain tank and the pressure in the RCS was at 1750 psi.
9:00 a.m.	The pressure in containment peaked at 4.5 psig.
9:00-11:00 a.m.	The RCS pressure increased from 1250 psi to 2100 psi.
11:30 a.m.	The operator opened the pressurizer relief-valve to depressurize the RCS in an attempt to initiate RHR cooling at 400 psi.
12:00 a.m.-1:00 p.m.	The RCS pressure decreased to about 500 psi and the core flooding tanks partially discharged. The relief capacity was not sufficient to vent enough to reach 400 psi.
2:00 p.m.	The pressure in the containment spikes at 28 psig causing containment sprays to be initiated. The operator stopped the spray pumps after about 2 minutes of operation.

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Time (approximate)	Discussion of Events
5:30 pm	The pressurizer relief valve was closed in order to repressurize the reactor coolant system.
5:30 - 8 pm	The RCS pressure increased from 650 psi to 2300 psi.
8 pm	RC pump in Loop A was started at which time the hot leg temperature decreased to about 560 degrees F and the cold leg temperature increased to 400 degrees F, indicating flow through the steam generator. Thereafter, the reactor was being cooled by reestablishing condenser vacuum and steaming to the condenser by steam generator A with the RCS cooled to about 280 degrees F and 1000 psi.

March 29

The RCS temperature and pressure was stabilized at about 280 degrees F and 840 to 1020 psi. The maximum reading on the incore thermocouples was 612^{o}F, but several thermocouples were not within range for computer readouts, i.e., the temperatures were higher than about 700 degrees F.

March 30

The RCS temperature and pressure were stable at about 280 degrees F and about 1000 to 1050 psi. Several incore thermocouples were beyond the range for computer readout, the maximum indicated reading was 659 degrees F. The licensee estimated the size of a bubble of non-condensible gas in the RCS to be about 1200 ft^3 at 875 psig.

March 31

The RCS temperature and pressure remained stable at about 280^{o}F and 1000 psi. Slight drop in pressurizer level 251-191". Temperatures in the core as measured from the incore thermocouples were gradually decreasing (maximum indicated about 500^{o}F). The hydrogen recombiner was in an operable status but additional shielding was needed and was being obtained. Two samples of containment atmosphere were analyzed which showed a hydrogen concentration of 1.7% and 1.0%. Licensee estimated the bubble size to be about 620 ft^3 @ 875 psig.

April 1

No substantial change in RCS temperature and pressure
Incore thermocouples continue to show decreased trend.

Licensee continued hookup of hydrogen recombiners and addition of shielding. Licensee calculated valves of bubble size varied. Containment air samples indicate 2.3% hydrogen.

April 2

Reactor pressure stable at about 1000 psi. Incore thermocouples continued to show a decrease with all measurements below 475°F. Inlet and outlet temperatures were still about 280°F. One hydrogen recombiner was put in operation to decrease the hydrogen gas concentration in the containment building.

Analysis indicated that the oxygen generation rate in reactor less than originally estimated. Measurements indicated that the bubble was being significantly reduced by degassing operations.

April 3

Reactor pressure and temperature stable at 1000 psi and 280°F, respectively. Thermocouple readings analyzed- maximum 477°F, only 3 thermocouples were above 400°F. Gas bubble size much reduced. Containment about 1.9% hydrogen. One pressurizer le[illegible]l indicator failed.

April 4

Reactor pressure and temperature stable at 1000 psi and 280°F, respectively. Thermocouple maximum temperature was 466°F. Gas bubble size decreasing. Vent valve on pressurizer intermittently open[illegible] and degassing continues through letdown system.

April 5

Reactor pressure and temperature s[illegible]ble at 1000 psi and 280°F, respectively. Maximum thermocouple reading is 462°F. Pressurizer level responding normally to pressure changes indicating a completely full system.

Containment atmosphere indicates 2% hydrogen. One recombiner operating, one in standby. Pressurizer vented to containment about 15 minutes every 6-8 hours.

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April 6

Reactor pressure stable at about 1000 psi and temperature about 285°F.

At approximately 1:25 pm, reactor coolant pump 1A tripped and reactor coolant pump 2A was started within about 2 minutes. Shift in thermocouple readings. The three thermocouples previously reading about 400°F are presently reading between 285°F and 315°F. Central thermocouple increased from 375°F to 425°F and is the only one now reading above 400°F.

Containment measurements indicate about 2% hydrogen. Pump-back system for pumping waste gas decay tank volume to containment began.

April 7

Reactor pressure and temperature stable at about 1000 psi and 280°F, respectively.

At about 8:00 pm, the licensee began to slowly lower reactor system pressure in increments of 50 psig. The slow decrease ended when reactor pressure reached 500 psi. This intentional pressure reduction expanded gasses trapped in control rod drive housings above the vessel head so that they could be dissolved or entrained and then be gassed through pressurizer venting and letdown at higher pressures. This degasification process is designed to prevent bubble formation as pressure and temperature decrease during the placement of the reactor cooling system in a long term, shutdown cooling mode.

Hydrogen concentration in the containment is about 1.7%.

THREE MILE ISLAND NUCLEAR STATION

TOUR

AND

BRIEFING

JANUARY 1978

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I

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I. Briefing Notes

A. NRC Inspection Program Purpose

To examine, on a representative sample basis, licensee, contractor, and vendor safety-related activities, controls, records, and work performance to ascertain compliance with license provisions and regulations relating to health, safety, common defense, security, and protection of the environment; establish a basis to issue, deny, suspend, modify or revoke a license; investigation of accidents or unusual circumstances involving facilities subject to licensing and regulation; and to document such inspection for official review and public dissemination.

B. Inspector/Licensee Interface

Inspections are usually unannounced, with contact made with the senior licensee representative upon arrival at the site. Findings are communicated during the Management Interview at the end of the inspection. Management Interview attendees normally include licensee and contractor site Managers. If there are serious findings, the Corporate Vice President responsible for the activity inspected usually is asked to attend. Inspection results are forwarded to the licensee, who is required to respond in writing to Items of Noncompliance. This response must include what the licensee has done or intends to do to correct Items of Noncompliance.

C. NRC Inspection Scope

There are basically six phases of inspection for reactors. These are the Pre-Construction Permit, Construction, Preoperational and Operating Preparedness, Startup, Operations and Decommissioning.

1. Pre-Construction Phase

The Pre-Construction Permit phase begins before the Construction Permit is issued. NRC inspectors conduct a substantive review of the prospective licensee's quality assurance program for early design and procurement prior to accepting an application for docketing. After the application is accepted and under review, site inspections are conducted to evaluate the applicant's site preparation program. During this phase,

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the frequency of inspection is dependent upon the amount and extent of work being performed by the applicant.

2. Construction Phase

The second inspection phase begins when the Construction Permit is issued and continues until all construction activities are complete. NRC inspectors examine, for safety-related activities, quality assurance program controls, documentation, and work performance. Typical items reviewed include electrical cabling, emergency power sources, containment building construction, reactor components, safety-related piping, welding practices and qualifications, nondestructive testing and procured item documentation.

3. Preoperational Testing and Operational Preparedness Phase

This phase begins approximately one year prior to completion of construction and extends through the issuance of an operating license. The minimum inspection frequency is eight per year although typically considerably more are conducted. The NRC inspectors verify through direct observation, personnel interviews, and review of facility records that: systems and components important to the safety of the plant are fully tested to demonstrate satisfaction of their design requirements; and management controls and procedures, including quality assurance programs, necessary for operation of the facility have been implemented and documented. Inspections are also conducted in the areas of plant staffing, training and emergency planning to ensure the facility is prepared to manage nuclear fuel.

4. Startup Phase

This inspection phase begins approximately six months before issuance of the operating license and continues until after the facility achieves full power and commercial operating status. NRC inspectors verify through direct observation, independent measurements, personnel interviews, and review of facility records that: systems and components important to the safety of the plant are tested under normal operating and abnormal transient conditions to demonstrate satisfaction with respect to their design requirements; and quality assurance and other

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management controls are implemented for the test program consistent with regulatory requirements.

5. Operations Phase

The operations phase is effective when an operating license has been issued. Fuel may be loaded into the reactor only when the operating license has been issued. The operations phase inspection program remains in effect for the life of the facility (about 40 years), that is until the facility license is significantly modified to reflect a deactivation, or decommissioned status. During the operational phase, NRC inspectors verify through direct observation, independent measurements, personnel interviews, and review of facility records and procedures that: the licensee's management control program is effective; and whether the facility is being operated safely and in conformance with regulatory requirements.

During this phase, four branches of the NRC's Regional office are involved in the inspection program. The Reactor Operations and Nuclear Support Branch, the Fuel Facility and Materials Safety Branch, and the Safeguards Branch carry out a uniform inspection program at the facility throughout its operational lifetime (about 40 years). The Reactor Construction Engineering Support Branch provides Engineering Support Specialists to provide additional expertise to other branches during the Operations Phase.

6. Decommissioning Phase

After the useful life of a facility and the facility is to be deactivated, the inspection program will verify that the licensee's program and controls to deactivate the facility are effective and in conformance with the license and the regulatory requirements covering this phase. Site inspections assure that facilities that have undergone or are undergoing decommissioning are properly maintained throughout the lifetime of this phase.

D. Inspection Frequency

Inspections at each facility are accomplished in accordance with a program established by the NRC Inspection and Enforcement Headquarters. The program is then transferred to an inspection plan for each

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facility. This inspection plan is augmented as necessary with special inspections. This approach ensures a degree of uniformity for inspections conducted all over the country. In 1977, Three Mile Island 1 had 37 inspections and Three Mile Island 2 had 45 inspections. In 1976, Three Mile Island 1 had 29 inspections and Three Mile Island 2 had 20 inspections. In 1975, Three Mile Island 1 had 28 inspections and Three Mile Island 2 had 15 inspections. In 1974, these plants had 37 and 9 inspections for Units 1 and 2, respectively. Since the start of construction, 203 inspections of Unit 1 and 106 inspections of Unit 2 have been conducted.

E. Inspector Training

It takes three to six months to qualify an inspector for unassisted site inspections and in excess of one (1) year to develop inspectors to the expected proficiency level. There are two inspector categories, Project Inspectors and Specialist Inspectors.

The Project Inspector has primary responsibility for the conduct of an appropriate and thorough inspection program. To accomplish this, he plans the inspection program, performs inspections, and monitors and coordinates inspections performed by Specialist Inspectors.

Specialist Inspectors are knowledgeable in given areas in which they focus their inspections. This assures high quality and uniform inspections in these areas for all facilities.

The combination of Project and Specialist Inspectors assures that a thorough, well-planned, uniform, and competent inspection program is conducted.

F. Staffing of Region I

Attachment A shows the organization of Region I. A Project Inspector is assigned to each reactor facility. Specialist Inspectors from the Radiological and Environmental Protection Branch, the Materials and Plant Protection Branch, and the Engineering and Nuclear Support Sections are also utilized at these facilities as required.

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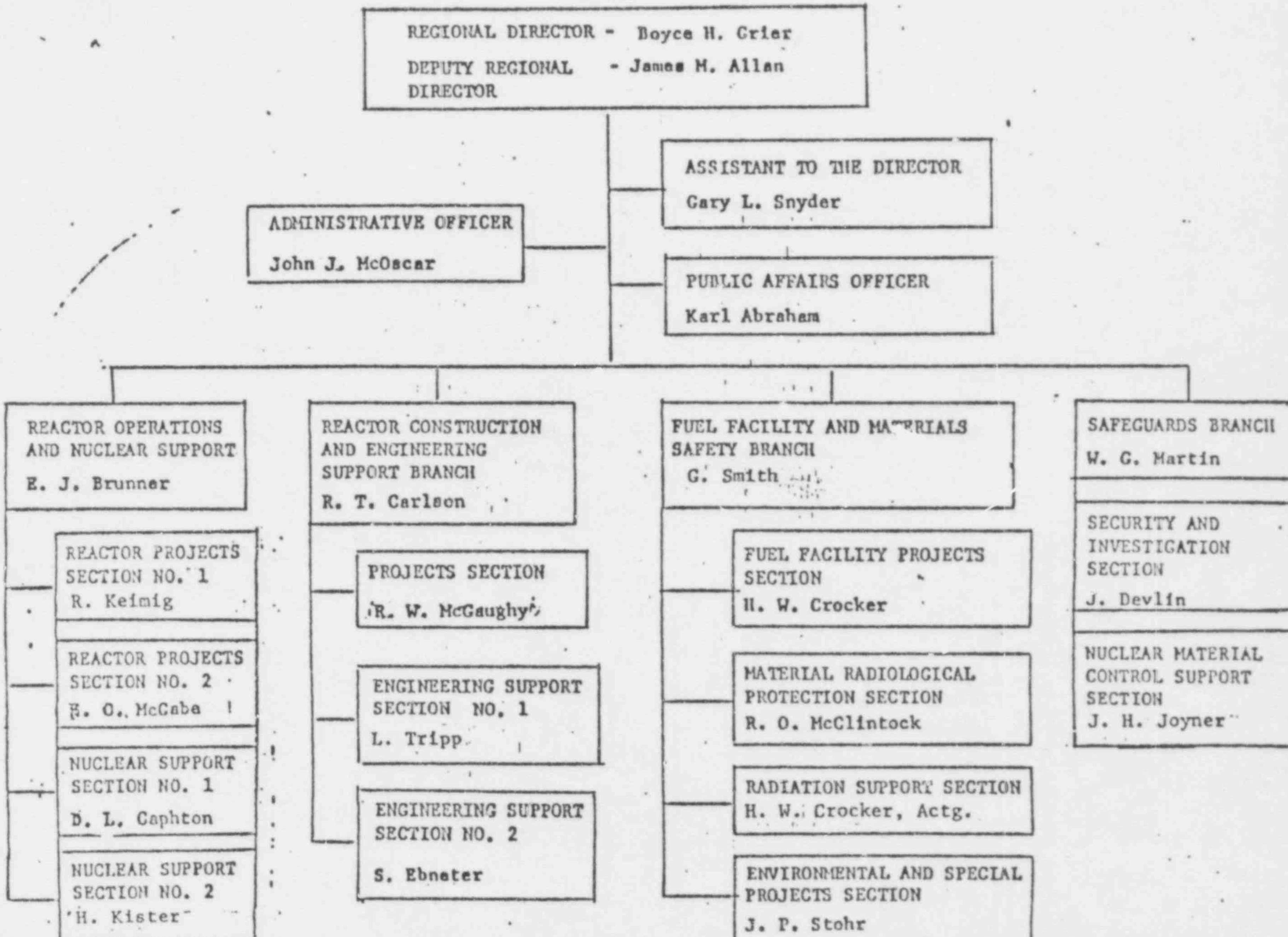
U. S. NUCLEAR REGULATORY COMMISSION

REGION I



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II. Attachments

A. General

Three Mile Island Nuclear Station consists of two units of the same design but with some minor variations between plants. The nuclear steam supply system for each unit is a two loop pressurized water reactor. Unit 1 is designed to operate at power levels up to 2535 MWt and will provide a gross electrical output of 871 MWe. Unit 2 is designed to operate at power levels up to 2772 MWt and will provide a gross electrical output of 959 MWe.

B. Ownership and Management

Three Mile Island Nuclear Station is owned jointly by three wholly owned subsidiary operating companies of General Public Utilities Corporation (GPU) in the following proportions:

Metropolitan Edison Company (Met Ed)	50%
Jersey Central Power & Light Company (Jersey Central)	25%
Pennsylvania Electric Company (Penelec)	25%

Met-Ed is the licensee and has complete responsibility for engineering, design, construction, operation and maintenance of the station.

The principal contractors who support Met-Ed in design, construction, testing and start-up of the station are:

Unit 1

Gilbert Associates, Inc. - Architect-Engineer
Babcock & Wilcox, Company - Nuclear Steam System Supplier
United Engineers & Constructors, Inc. - Construction Management
United Engineers & Constructors, Inc. - Principal Construction Contractor

Unit 2

Burns and Roe, Inc. - Architect-Engineer
Babcock & Wilcox Company - Nuclear Steam System Supplier
United Engineers & Constructors, Inc. - Construction Manager
United Engineers & Constructors, Inc. - Principal Construction Contractor
GPU Service Company - On-Site Owners Agent

In addition to acting as owner's agent for construction, GPU Service Company is responsible for Quality Assurance.

C. History

Unit 1

The initial application for Three Mile Island Nuclear Station was submitted by Met-Ed as the sole owner on April 28, 1967, as an application for a Construction Permit and an operating license for a single unit plant. A provisional Construction Permit for this plant was granted on May 18, 1968.

An application for an Operating License for Unit 1 was submitted on June 25, 1970, and was amended on January 12, 1971, to reflect the joint ownership by Met-Ed, Jersey Central, and Penelec.

Construction of Unit 1 started in January 1968, and was completed in March 1974. The Operating License was issued on April 19, 1974. Commercial operation began on September 2, 1974. The unit capacity factor through November 1977, is 73.6%.

Unit 2

The application for a construction permit for Unit 2 was initially submitted for a location in New Jersey as Oyster Creek Unit 2 by Jersey Central early in 1968. The site was transferred to Three Mile Island and the unit designated as Three Mile Island 2 in February 1969. The application was also revised to identify Met-Ed and Jersey Central jointly as the applicant. An exemption authorizing start of some construction was authorized on June 27, 1969, and a provisional construction permit was granted on November 4, 1969.

Construction was started on July 1, 1969, and was about 83% complete as of July 1, 1975, and was about 97% complete as of December 1977. The presently scheduled fuel loading date is March 1978.

A revised application for an operating license was submitted on April 4, 1974, and is still under review by Licensing. This application was submitted by Met-Ed as the proposed licensee and identifies Met-Ed, Jersey Central, and Penelec as the joint owners.

D. Description of Three Mile Island Nuclear Station

1. Three Mile Island Station Site

The site for the Three Mile Island Station is located in Londonderry Township, Dauphin County, in Southeastern Pennsylvania, on Three Mile Island near the East Shore of the Susquehanna River. The site is about 70 miles northeast of Baltimore, Maryland, 120 miles west of Philadelphia, Pennsylvania, and 5 miles southeast of Harrisburg, Pennsylvania.

Three Mile Island is about 11,000 feet long and 17[illegible] feet wide. It is located about 900 feet from the east bank of the Susquehanna River with its longer axis approximately due north and south and parallel to the flow of the river. The generating station, consisting of two units is located on the northern one-third of the island.

Metropolitan Edison Company owns Three Mile Island in its entirety as well as all but several acres of Shelley Island adjacent to it on the west. The exclusion area is 2,000 feet in radius and includes portions of Three Mile Island, the river surface around it and a small piece of Shelley Island in the portion owned by Metropolitan Edison Company.

Plant grade (Elev. 304) is above the water surface elevation for the "design" flood. Protective dikes are provided to protect the plant from wave action during a "design" flood. At the northern tip of the island the top of the dike is at Elev. 310. It slopes down uniformly to Elev. 305 south of the site with a protective dike across the island at this location at Elev. 304.

In the event of a "Probable Maximum Flood" (PMF), the dikes will overtopped and flood waters will reach the station buildings. The buildings are protected against flooding in that case up to a level of 4 feet above the PMF water level.

2. Plant Layout

As shown on the attached plot plan, the units are located at the northern end of Three Mile Island and a[illegible]t to each other. The Nuclear Steam Systems are ins[illegible]d

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within the containment buildings with the fuel handling buildings between them. The control buildings are southeast and east of Units 1 and 2 respectively. The turbine buildings are east of Unit 1 and south of Unit 2, while the cooling towers are to the northeast and southeast. The river water intakes are located adjacent to each other on the west bank of the island.

Certain facilities such as the electrical switchyard, new fuel storage, fuel handling crane, Demineralized water storage, etc. are shared by Units 1 and 2. However, none of the shared facilities affect the operation of the nuclear steam supply system, the engineered safety features or the controls of either units.

3. Nuclear Steam Supply System

Each nuclear steam supply system includes the reactor core and core supports housed within the reactor pressure vessel; the reactor control system; the reactor cooling system including the reactor vessel, four reactor coolant pumps, two steam generators, a pressurizer, interconnecting piping and valves; together with instrumentation, controls and their interconnecting wiring and piping.

a. Reactor Core

The reactor core is made up of 177 fuel assemblies arranged in a square lattice approximating a cylindrical shape. The entire core is supported on and maintained in position by a stainless steel structural cage, the reactor internals, which also serves to direct the flow within the pressure vessel. The fuel is composed of low enriched uranium dioxide encased in zircaloy-4 tubing. 208 of these rods are assembled in a 15 x 15 array into each of the fuel assemblies which also provides space for entry of 16 control rods and incore instrumentation. Control of reactivity is provided by control rod assemblies, burnable poisons and soluble boron in the cooling water. The control rods are an alloy of silver-indium-cadmium encased in stainless steel tubing. Sixty-one (61) control rod assemblies are provided and are positioned by an electric motor driven mechanical drive. The position of each control assembly is indicated in the reactor control room. The burnable poison is composed of Aluminum Oxide-Boron Carbide encased in Zircaloy 4.

b. Reactor Pressure Vessel

The reactor vessel is approximately 15 feet in diameter and has an overall height of about 40 feet. The minimum shell thickness is 8-7/16 inches. The total weight of the vessel and head is over 440 tons. It is made of Carbon Steel with an internal stainless steel cladding approximately 3/16" thick.

c. Reactor Cooling System

The reactor cooling system operates at a design pressure and temperature of 2500 psig and 650°F. The reactor coolant pumps are single stage, vertical, centrifugal pumps. Two pumps are provided on each loop of the system each with a capacity of 92,000 gpm at 362 feet head. They are rated at 9,000 horsepower and including the motor weigh approximately 60 tons apiece.

The steam generators are a once through shell and tube design, approximately 12 feet in diameter and 73 feet in height with a minimum shell thickness of 4-3/16 inches. At full load conditions they will produce steam at 910 psig and 570°F. One steam generator is provided on each loop.

A single pressurizer is provided in order to maintain pressure in the reactor coolant system and prevent boiling. It is approximately 8 feet in diameter and 45 feet in height with a minimum shell thickness at 6-3/16 inches. Electric heaters are provided in the lower section and a water spray nozzle in the upper head in order to control the pressure and temperature.

Reactor piping is carbon steel clad on the inner face with stainless steel or inconel. The reactor inlet piping is 28 inches in diameter and the reactor outlet piping is 36 inches in diameter.

4. Containment

The reactor building is a reinforced concrete cylindrical structure with a dome roof on a flat foundation mat. The cylindrical wall and dome roof are prestressed using a post tensioning system. The building is steel-lined for containment. The liner plate is 1/4 inch to 1/2 inch thick and designed as a leak-tight membrane. The building is 130 feet in diameter and 157 feet in height to the spring line of the dome. The walls are 4 feet thick and the dome is 3 feet thick. The foundation mat bears on rock and is 11 feet 6 inches thick with a 2 feet slab over the base liner plate.

A 23 feet diameter equipment hatch and two 9 feet diameter personnel air locks, one of them within the equipment hatch, are provided. All penetrations of the building; piping, electrical, etc. are carried through leak tight penetrations assemblies welded to the building liner with isolation valves provided on all piping. The reactor building has been designed for an internal pressure of 60 psig and will be pressure tested at 69 psig.

Because of its proximity to Harrisburg International Airport, the Category I buildings have been designed to withstand aircraft impact loadings based on a 200,000 lb aircraft at a velocity of 200 knots.

5. Engineered Safety Features

Engineered safety features are provided so that under the most severe loss of coolant accident (LOCA) they will prevent meltdown of the core, maintain the integrity of the containment and ensure that public exposure to radiations will be below the 10 CFR 100 limits. They are provided with redundant equipment and power sources so as to remain effective under such conditions.

The following systems are provided:

a. Containment Systems including:

(1) Containment structure described above.

(2) Containment heat removal systems consisting of the reactor building spray system and the reactor building air cooling system. The air cooling system consists of five water to air cooling units. This system functions during normal operation using three units. Under LOCA conditions all five units are operated to prevent an increase in the air temperature

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supplied from the borated water storage tank or alternately may be recirculated from the reactor sump. This system consists of two independent circuits either of which will satisfy the requirements imposed by a LOCA. This system is normally operated as part of the decay heat removal system during shutdown of the reactor.

6. Support Systems

Although not classed as engineered safety features systems, the following systems are required for their support.

a. Emergency power
b. Nuclear services river water system
c. Nuclear services closed cooling water system
d. Decay heat closed cooling water system

7. Instrumentation and Controls

Instrumentation and controls include the safety related systems as well as non-safety related systems which are necessary for effective operation of the plant. The safety related systems include:

a. The reactor protection system which serves to shutdown the reactor by tripping the control rods when required.

b. The safety features actuation system which initiates control signals so as to energize the engineered safety features systems described above.

All of the controls and display instrumentation necessary to startup, operate and shutdown the plant is located in the control room together with instruments necessary to maintain safe shutdown conditions after a LOCA.

All safety related wiring, cable trays, instrument mounts, etc. are color coded. Redundant lines are physically separated or if this is not possible are separated by barriers.

8. Electric Power

Power is generated at 22 Kv and stepped up to 500 Kv for transmission to a 500 Kv substation off-site.

Emergency on-site power is provided by two standby diesel generators rated at 3300 Kw each. The worst case load on the diesel generator is 2789 Kw thereby giving a safety factor of over 500 Kw or about 15%. The diesel generator set is utilized solely as a standby power supply service and it does not serve a secondary purpose such as power generation for peak demand periods, etc.

The normal onsite power system for the Three Mile Island Nuclear Station has been designed so that the electrical power from the transmission network is supplied by two physically independent circuits to minimize the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. In the event of a failure of either circuit, transfer of the loads to the remaining source is accomplished automatically. This concept is a basic part of the reliability analysis program performed using fault analysis as a design tool.

D-C power for controls, instrumentation and d-c motors is provided by two physically independent 250/125 volt circuits. Each circuit is supplied by rectifiers with batteries as an alternate source.

9. Auxiliary Systems

a. New Fuel

New fuel storage is provided in a separate and protected area for dry storage in the fuel storage area. This area is shared between Units 1 and 2.

b. Spent Fuel

Spent fuel is stored in two adjacent pools in the Fuel Handling Building, one for each unit. The pools are of reinforced concrete and lined with stainless steel. Pool water is borated and is circulated through a cooling system to remove decay heat. Cleanup is provided by bypass flow through a filter and demineralized.

c. Nuclear Services River Water System

This system provides cooling water for nuclear closed cycle systems, emergency diesel generator

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engine cooling, HVAC coolers in the control and services building, reactor building coolers, and also serves as a backup supply to the Nuclear Services Closed Cooling Water System. The system has 2 circuits each with 100% capacity for a LOCA.

d. Decay Heat Removal (DH) System

This system removes decay heat from the core during cooldown of the reactor. Two independent 100% capacity circuits are provided.

e. Decay Heat Closed Cooling Water System

Provides secondary cooling water to the DH system. Two independent 100% capacity circuts are provided.

f. Nuclear Services Closed Cooling Water System

Provides cooling water for nuclear equipment coolers, instrument air compressors and aftercoolers. Two 100% capacity circuits are provided with three 100% capacity pumps.

g. Intermediate Closed Cooling Water System

Provides cooling water to components in the reactor building which handles reactor coolant.

h. Secondary Services Closed Cooling Water System

Provides cooling water to non-nuclear related equipment.

i. Make-up, Purification and Chemical Addition System

This system controls and maintains the reactor coolant inventory and also controls the boron concentration in the reactor coolant system.

10. Service Systems

The following systems are provided as support services:

a. Demineralized water system
b. Portable and Sanitary Water

c. Compressied air systems
d. Equipment and floor drainage
e. Heating, ventilation and air conditioning systems
f. Fire Protection System
g. Communications System

11. Steam and Power Conversion System

The steam from each of the steam generators is supplied by two lines to a tandem compound two-stage reheat four-flow turbine which generates the electric power. Steam is supplied at 900 psia and 566°F to the high pressure turbine. Exhaust from this turbine enters four moisture separators for removal of excess moisture and reheat of the steam before entry into two low pressure turbines. Exhaust steam from the low pressure turbines is condensed, deaerated and collected in a dual pressure main surface condenser for recirculation by condensate and condensate booster pumps.

12. Radioactive Waste

Radioactive waste systems include liquid, gaseous and solid waste systems.

a. Liquid Waste Systems

The liquid waste system for each unit is housed within the Auxiliary Building. It consists of tanks, filters, evaporators, coolers and pumps with associated piping, valves instruments and controls. The liquid waste system consists of two chains; one to service reactor coolant waste and the other to service miscellaneous wastes.

(1) The reactor coolant waste chain provides for boron removal from reactor coolant, collection, holdup and processing by ion exchange to remove fission products, recovery of boric acid, volume reduction by evaporation, stripping of radioactive gases, recycle of distillate for reuse and transfer of concentrated waste for disposal.

(2) The miscellaneous waste chain collects liquid wastes from drains, demineralizer regeneration

laundry and shower processes the waste by evaporation at Unit 1 and disposes material as required.

b. Gaseous Waste Systems

Gases are normally compressed and stored in decay tanks. After decay to acceptable levels the contents of the tanks are sampled and analyzed for activity. If acceptable the tanks are discharged through a filter train. All releases are manually controlled and monitored.

c. Solid Waste Systems

Solid wastes consists of liquid waste concentrations and spent demineralizers resins in addition to the normal solid material which has become contaminated. Liquid waste concentrates are stored in the concentrated liquid waste storage tanks which are heated and covered with a nitrogen blanket. Spent resins are allowed to decay and then transferred to the spent resin storage tank at Unit 1. Other waste material is packaged, placed in 55 gallon drums and stored at Unit 1.

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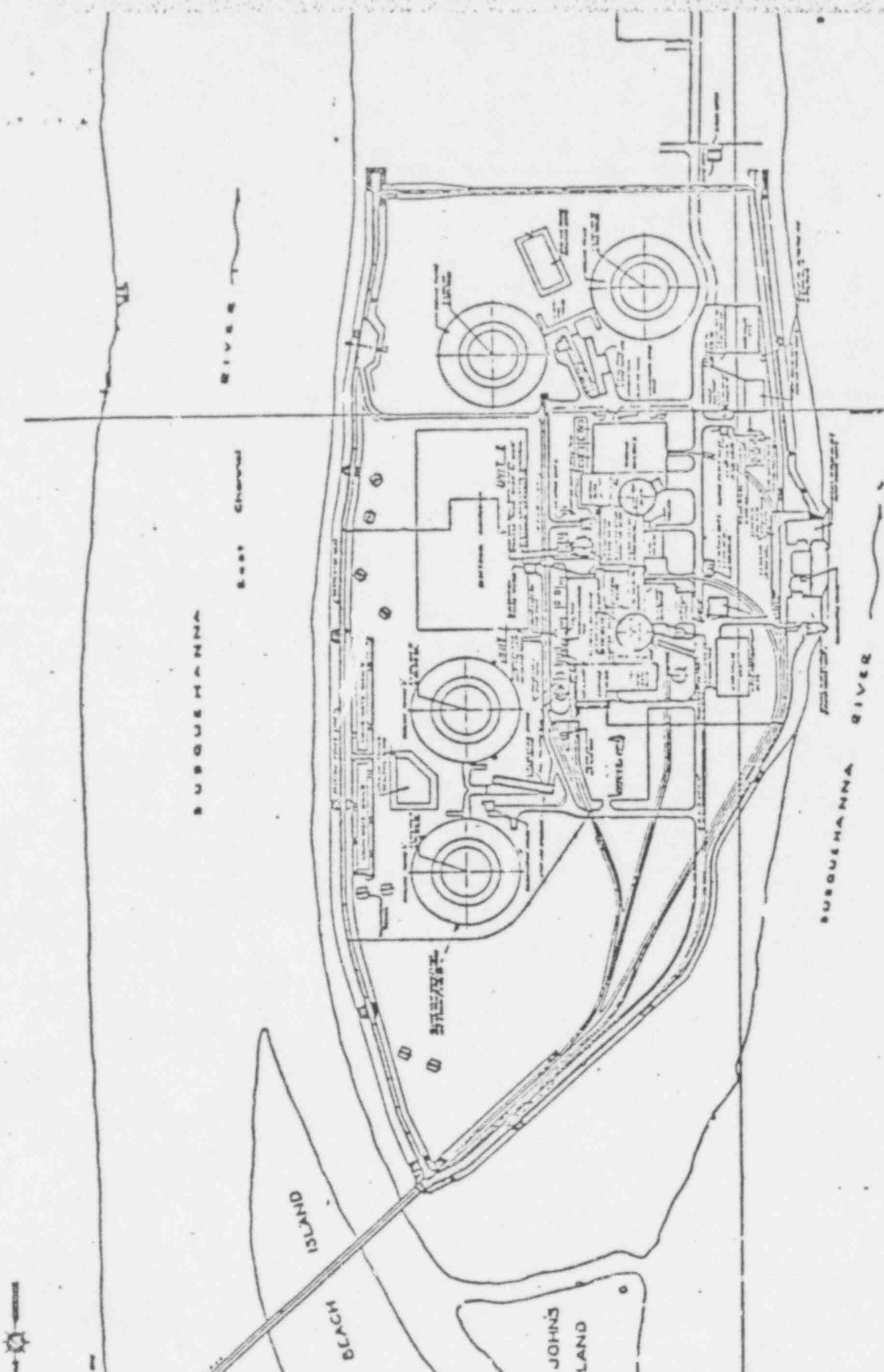

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