ATTACHMENT 1

CORRECTIONS TO THE FINAL DRAFT RETYPED
NINE MILE POINT UNIT 1
TECHNICAL SPECIFICATIONS

Page No.	Corrections
Index	Bases are not included in the index
2-5	See Amendment 92 for revised Figure 2.1.1 Flow Biased Scram and APRM Rod Block
B2-2	missing period after second last sentence
B2-3	change "Annunciation" to "annunciation"
B2-8	See Amendment 92 for revised paragraph F
3/4 1-21	missing punctuation
3/4 1-31	See Amendment 92 for revised Figure 3.1.7 aa Limiting Power Flow Line
3/4 2-12	change "conductvity" to "conductivity"
3/4 2-14	change "very" to "every"
3/4 4-8	change specification 4.4.4.g as shown in Amendment 91
3/4 4-9	change "Specifiction" to "Specification"
3/4 6-16	change "SHTUDOWN" to "SHUTDOWN"
3/4 6-43	delete "NOTES FOR TABLES"
3/4 6-46	change "Undervolt" to "Undervoltage"
3/4 6-83	delete extra comma
3/4 6-100	change "indictor" to "indicator"
3/4 6-134	missing "of"
3/4 6-147	change "(pCi/m)" to (pCi/m^3)"
B3/4 1-1	change "met" to "meet"
B3/4 1-18	see Amendment 92 for revised paragraph on Power/Flow relationship
B3/4 1-19	change "blow" to "below"
B3/4 1-21	see Amendment 92 for reference number 14
B3/4 3-10	missing "operation"
B3/4 6-22	missing "is"
6-4	see Amendment 89 for revised Figure 6.2-1, Station Management Organization Chart
6-5	See NMPC application for amendment dated February 20, 1987, for revised Figure 6.2-2, Site Operation Organization
6-6	change "Oeprators" to "Operators"

Because the boiling transition correlation is based on a large quantity of full scale data there is a very high confidence that operation of a fuel assembly at the condition of the SLCPR would not produce boiling transition. Thus, although it is not required to establish the safety limit, additional margin exists between the safety limit and the actual occurrence of loss of cladding integrity.

However, if boiling transition were to occur, clad perforation would not be expected. Cladding temperatures would increase to approximately 1100°F which is below the perforation temperature of the cladding material. This has been verified by tests in the General Electric Test Reactor (GETR) where similar fuel operated above the critical heat flux for a significant period of time (30 minutes) without clad perforation.

If reactor pressure should ever exceed 1400 psia during normal power operating (the limit of applicability of the boiling transition correlation) it would be assumed that the fuel cladding integrity safety limit has been violated.

In addition to the boiling transition limit SLCPR operation is constrained to a maximum LHGR of 13.4 kW/ft for 8x8 fuel and 13.4 kW/ft for 8x8R fuel. At 100% power this limit is reached with a Maximum Total Peaking Factor (MTPF) of 3.02 for 8x8 fuel and 3.00 for 8x8R fuel. For the case of the MTPF exceeding these values, operation is permitted only at less than 100% of rated thermal power and only with reduced APRM scram settings as required by Specification 2.1.2.a. (In cases where for a short period the total peaking factor was above 3.02 for 8x8 fuel and 3.00 for 8x8R fuel the equation in Figure 2.1.1 will be used to adjust the flow biased scram and APRM rod block set points.)

At pressure equal to or below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low power and all core flows, this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low powers and all flows will always be greater than 4.56 psi.

Analyses show that with a bundle flow of 28×10^3 lb/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Therefore, due to the 4.56 psi driving head, the bundle flow will be greater than 28×10^3 lb/hr irrespective of total core flow and independent of bundle power for the range of bundle powers of concern. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at 28×10^3 lb/hr is approximately 3.35 MWt. With the design peaking factor, this corresponds to a core thermal power of more than 50%. Thus, a core thermal power limit of 25% for reactor pressures below 800 psia or core flow less than 10% is conservative.

During transient operation the heat flux (thermal power-to-water) would lag behind the neutron flux due to the inherent heat transfer time constant of the fuel which is 8 to 9 seconds. Also, the limiting safety system scram settings are at values which will not allow the reactor to be operated above the safety limit during normal operation or during other plant operating situations which have been analyzed in detail. (3,4) In addition, control rod scrams are such that for normal operating transients the neutron flux transient is terminated before a significant increase in surface heat flux occurs. Scram times of each control rod are checked periodically to assume adequate insertion times. Exceeding a neutron flux scram setting and a failure of the control rods to reduce flux to less than the scram setting within 1.5 seconds does not necessarily imply that fuel is damaged; however, for this specification a safety limit violation will be assumed any time a neutron flux scram setting is exceeded for longer than 1.5 seconds.

If the scram occurs such that the neutron flux dwell time above the limiting safety system setting is less than 1.7 seconds, the safety limit will not be exceeded for normal turbine or generator trips, which are the most severe normal operating transients expected. These analyses show that even if the bypass system fails to operate, the design limit of the SLCPR is not exceeded. Thus, use of a 1.5-second limit provides additional margin.

The process computer has a sequence annunication program which will indicate the sequence in which scrams occur such as neutron flux, pressure, etc. This program also indicates when the scram set point is cleared. This will provide information on how long a scram condition exists and thus provide some measure of the energy added during a transient. Thus, computer information normally will be available for analyzing scrams; however, if the computer information should not be available for any scram analysis, Specification 2.1.1.c will be relied on to determine if a safety limit has been violated.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
is greater than 110 psig and the reactor coolant temperature is greater than saturation temperature, all six solenoid-actuated pressure relief valves shall be operable.	valve shall be manually opened until acoustic monitors or thermocouples downstream of the valve indicate that the valve has opened and steam is flowing from the valve.
b. If Specification 3.1.5a above is not met, the reactor coolant pressure and the reactor coolant temperature shall be reduced to 110 psig or less and saturation temperature or less, respectively, within ten hours.	b. At least once during each operating cycle, automatic initiation shall be demonstrated.

c. The limits specified in 3.2.3a and 3.2.3b may be exceeded for a period of time not to exceed 24 hours. In no case shall (1) the conductivity exceed a maximum limit of 10 μmho/cm, or (2) the chloride ion concentration exceed a maximum limit of 0.5 ppm.

d. If Specifications 3.2.3.a, b, and c are not met, normal orderly shutdown shall be initiated within one hour and the reactor shall be in the cold shutdown condition within ten hours.

e. If the continuous conductivity monitor is inoperable for more than 7 days the reactor shall be placed in the cold shutdown condition within 24 hours.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.2.5 REACTOR COOLANT SYSTEM LEAKAGE

Applicability:

Applies to the limits on reactor coolant system leakage rate and leakage detection systems.

Objective:

To assure that the makeup capability provided by the control rod drive pump is not exceeded.

Specification:

a. Any time irradiated fuel is in the reactor vessel and the reactor temperature is above 212°F, reactor coolant leakage into the primary containment shall be limited to:

   1. Five gallons per minute unidentified leakage.

   2. A two gallon per minute increase in unidentified leakage within any period of 24 hours or less.

   3. Twenty-five gallons per minute total leakage (identified plus unidentified) averaged over any 24 hour period.

4.2.5 REACTOR COOLANT SYSTEM LEAKAGAE

Applicability:

Applies to the monitoring of reactor coolant system leakage.

Objective:

To determine the reactor coolant system leakage rate and assure that the leakage limits are not exceeded.

Specification:

a. A check of the reactor coolant leakage shall be made very four hours.

every

3.4.5 CONTROL ROOM AIR TREATMENT SYSTEM

Applicability:

Applies to the operating status of the control room air treatment system.

Objective:

To assure the capability of the control room air treatment system to minimize the amount of radioactivity or other gases entering the control room in the event of an incident.

Specification:

a. Except as specified in Specification 3.4.5e below, the control room air treatment system and the diesel generators required for operation of this system shall be operable at all times when containment integrity is required.

b. The results of the in-place cold DOP and halogenated hydrocarbon test design flows on HEPA filters and charcoal adsorber banks shall show ≥99% DOP removal and ≥99% halogenated hydrocarbon removal when tested in accordance with ANSI N.510-1980.

4.4.5 CONTROL ROOM AIR TREATMENT SYSTEM

Applicability:

Applies to the testing of the control room air treatment system.

Objective:

To assure the operability of the control room air treatment system.

Specification:

a. At least once per operating cycle, or once every 24 months, whichever occurs first, the pressure drop across the combined HEPA filters and charcoal adsorber banks shall be demonstrated to be less than 6 inches of water at system design flow rate (±10%).

b. The tests and sample analysis of Specification 3.4.5b, c and d shall be performed at least once per operating cycle or once every 24 months, or after 720 hours of system operation, whichever occurs first or following significant painting, fire or chemical release in any ventilation zone communicating with the system.

TABLE 3.6.2b (Cont'd)

INSTRUMENTATION THAT INITIATES
PRIMARY COOLANT SYSTEM OR CONTAINMENT ISOLATION

Limiting Condition for Operation

Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels Per Operable Trip System	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable			
				Shutdown	Refuel	Startup	Run
CLEANUP SYSTEM ISOLATION							
(8) High Area Temperature	1	2	≤190	x	x	x	x
SHUTDOWN COOLING SYSTEM ISOLATION							
(9) High Area Temperature	1	1	≤170	x	x	x	x
CONTAINMENT ISOLATION							
(10) Low-Low Reactor Water	2	2	≥5 inches (Indicator Scale)	(c)		x	x

TABLE 4.6.2h

OFFGAS AND VACUUM PUMP ISOLATION

Surveillance Requirement

Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
OFFGAS			
(1) High Radiation Offgas Line			
a. Upscale	Once/shift	Once per week	Once per 3 months
b. Downscale	Once/shift	Once per week	Once per 3 months
VACUUM PUMP			
(2) High Radiation Main Steam Line	Once/shift	Once per week	Once per 3 months

NOTES FOR TABLES 3.6.2h and 4.6.2h

Table 3.6.2i (Cont'd)

DIESEL GENERATOR INITIATION

Limiting Condition for Operation

Parameter	Set Point (Inverse Time Undervoltage Relays)	
Loss of Power	Relay Dropout	Operating Time[a]
a. 4.16kV PB 102/103 Emergency Bus Undervolt~~-~~(age) (Loss of Voltage)	≥3200 volts	0 volts ≤3.2 seconds
b. 4.16kV PB 102/103 Emergency Bus Undervoltage (Degraded Voltage)	≥3600 volts	3580 volts 18.5 ± 3 seconds

(a) The operating time indicated in the table is the time required for the relay to operate its contacts when the voltage is suddenly decreased from operating voltage level values to the voltage level listed in the table above.

3.6.8 CARBON DIOXIDE SUPPRESSION SYSTEM

Applicability:

Applies to the operational status of the carbon dioxide suppression system.

Objective:

To assure the capability of the carbon dioxide suppression system to provide fire suppression in the event of a fire.

Specification:

a. The CO_2 system, which supplies the Recirculation Pumps Motor-Generator Sets, Power Boards 102 and 103, Diesel Generators 102 and 103, Cable Room fire hazards, shall be OPERABLE with a minimum level of 40% of tank and a minimum pressure of 250 psig in the storage tank.

4.6.8 CARBON DIOXIDE SUPPRESSION SYSTEM

Applicability:

Applies to the periodic surveillance requirements of the carbon dioxide suppression system.

Objective:

To verify the operability of the carbon dioxide suppression system.

Specification:

a. The CO_2 system shall be demonstrated operable.

1. At least once per 7 days by verifying the CO_2 storage tank level and pressure.

2. At least once per 31 days by verifying that each valve, manual, power operated or automatic, in the flow path is in its correct position.

3. At least once every six months by verifying the system valves and associated ventilation dampers actuate automatically to a simulated actuation signal. A brief flow test shall be made to verify flow from each nozzle ("Puff Test").

TABLE 4.6.11

ACCIDENT MONITORING INSTRUMENTATION

SURVEILLANCE REQUIREMENT

Parameter	Instrument Channel Test	Instrument Channel Calibration
(1) Relief valve position indicator (Primary - Acoustic)	Once per month	Once during each major refueling outage
Relief valve position indicator (Backup - Thermocouple)	Once per month	Once during each major refueling outage
(2) Safety valve position indictor (Primary - Acoustic)	Once per month	Once during each major refueling outage
Safety valve position indicator (Backup - Thermocouple)	Once per month	Once during each major refueling outage
(3) Reactor vessel water level	Once per month	Once during each major refueling outage
(4) Drywell Pressure Monitor	Once per month	Once during each major refueling outage
(5) Suppression Chamber Water Level Monitor	Once per month	Once during each major refueling outage
(6) Containment Hydrogen Monitor	Once per month	Once per quarter
(7) Containment High Range Radiation Monitor	Once per month	Once during each major refueling outage
(8) Suppression Chamber Water Temperature	Once per month	Once during each major refueling outage

3.6.16 RADIOACTIVE EFFLUENT TREATMENT SYSTEMS (Cont'd)

b. Gaseous (Cont'd)

prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.3, Special Report that identifies the inoperable equipment and the reason for its inoperability, actions taken to restore the inoperable equipment to OPERABLE status, and a summary description of those actions taken to prevent a recurrence.

c. Solid

The solid radwaste system shall be used in accordance with a Process Control Program to process wet radioactive wastes to meet shipping and burial ground requirements.

With the provisions of the process control program not satisfied, suspend shipments of defectively processed or defectively packaged solid radioactive wastes from the site.

4.6.16 RADIOACTIVE EFFLUENT TREATMENT SYSTEMS (Cont'd)

c. Solid

The process control program shall be used to verify the solidification of at least one representative test specimen from at least every tenth batch of each type of wet radioactive waste (e.g., filter sludges and evaporator bottoms).

(1) If any test specimen fails to verify solidification, the solidification of the batch may then be resumed using the alternative solidification parameters determined by the process control program.

(2) If the initial test specimen from a batch of waste fails to verify solidification, the process control program shall provide for the collection and testing of representative test specimens from

TABLE 4.6.20-1

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS (a, b)
LOWER LIMIT OF DETECTION LLD (c)

Surveillance Requirement

Analysis	Water (c) (pCi/l)	Airborne Particulate or Gases (pCi/m^3)	Fish (pCi/kg, wet)	Milk (pCi/l)	Food Products (pCi/kg, wet)	Sediment (pCi/kg, dry)
gross beta	4	0.01				
H-3	3000					
Mn-54	15		130			
Fe-59	30		260			
Co-58, Co-60	15		130			
Zn-65	30		260			
Zr-95, Nb-95	15					
I-131	(d)	0.07		1	60	
Cs-134	15	0.05	130	15	60	150
Cs-137	18	0.06	150	18	80	180
Ba/La-140	15			15		

a. Reactivity Limitations

(1) Reactivity margin - core loading

The core reactivity limitation is a restriction to be applied to the design of new fuel which may be loaded in the core or into a particular refueling pattern. Satisfaction of the limitation can only be demonstrated at the time of loading or reloading and must be such that it will apply to the entire subsequent fuel cycle. It is sufficient that the core in its maximum reactivity condition be subcritical with the control rod of highest worth fully withdrawn and all other rods fully inserted. In order to implement this requirement, it will be required that the amount of shutdown margin will be at least R + 0.25 percent Δk in the cold, xenon-free condition. In this generalized expression the value of R is the difference between the calculated value of core reactivity anytime later in the cycle where it may be greater than at the beginning. R must be a positive quantity or zero. A core which contains temporary control curtains or other burnable neutron absorbers may have a reactivity characteristic which increases with core lifetime, goes through a maximum, and then decreases thereafter.

The 0.25 percent Δk in the expression R + 0.25 percent Δk is provided as a finite, demonstrable, subcriticality margin. For the first fuel cycle, core reactivity is calculated never to be greater than the beginning-of-life value; hence, R = 0. The new value of R must be determined for each fuel cycle.

(2) Reactivity margin - stuck control rods

The specified limits provide sufficient scram capability to accommodate failure to scram of any one operable rod. This failure is in addition to any inoperable rods that exist in the core, provided that those inoperable rods met the core reactivity Specification 3.1.1.a(1).
meet

Control rods which cannot be moved with control rod drive pressure are indicative of an abnormal operating condition on the affected rods and are, therefore, considered to be inoperable. Inoperable rods are valved out of service to fix their position in the core and assure predictable behavior. If the rod is fully inserted and then valved out of service, it is in a safe position of maximum contribution to shutdown reactivity. If it is valved out of service in a non-fully inserted position, that position is required to be consistent with the shutdown reactivity limitation stated in Specification 3.1.1 a(1), which assures the core can be shut down at all times with control rods.

Partial Loop Operation

The requirements of Specification 3.1.7e for partial loop operation in which the idle loop is isolated precludes the inadvertent startup of a recirculation pump with a cold leg. However, if these conditions cannot be met, power level is restricted to 90.5 percent power based on current transient analysis (Reference 9). For three loop operation power level is restricted to 90 percent power based on the Reference 13 LOCA analysis.

The results of the ECCS calculation are affected by one or more recirculation loops being unisolated and out of service. This is due to the fact that credit is taken for extended nucleate boiling caused by flow coastdown in the unbroken loops. The reduced core flow coastdown following the break results in higher peak clad temperature due to an earlier boiling transition time. The results of the ECCS calculations are also affected by one or more recirculation loops being isolated and out of service. The mass of water in the isolated loops unavailable during blowdown results in an earlier uncovery time for the hot node. This results in an increase in the peak clad temperture.

To assure peak clad temperatures remain blow 2200°F, analysis has shown that the limiting average planar linear heat generation rate for each fuel type shall be reduced 2 percent and 4 percent for 4 and 3 loop operation respectively (Reference 13).

Partial loop operation and its effect on lower plenum flow distribution is summarized in Reference 11. Since the lower plenum hydraulic design in a non-jet pump reactor is virtually identical to a jet pump reactor, application of these results is justified. Additionally, non-jet pump plants contain a cylindrical baffle plate which surrounds the guide tubes and distributes the impinging water jet and forces flow in a circumferential direction around the outside of the baffle.

Recirculation Loops

Requiring the suction and discharge for at least two (2) recirculation loops to be full open assures that an adequate flow path exists from the annular region between the pressure vessel wall and the core shroud, to the core region. This provides for communication between those areas thus assuring that reactor water level instrument readings are indicative of the water level in the core region.

When the reactor vessel is flooded to the level of the main steam line nozzle, communication between the core region and annulus exists above the core to ensure that indicative water level monitoring in the core region exists. When the steam separators and dryer are removed, safety limit 2.1.1d and e requires water level to

For reactor coolant temperatures less than 215°F not enough steam is generated during a loss-of-coolant accident to pressurize the containment. In fact, for coolant temperatures up to 312°F, the resultant loss-of-coolant accident pressure would not exceed the design pressure of 35 psig.

Operation of only one containment spray pump is sufficient to provide the required containment spray flow. The specified flow of 3000 gpm (approximately 95 percent to the drywell and the balance to the suppression chamber) is sufficient to remove post-accident core energy released including a substantial chemical reaction involving hydrogen generation and will also limit pressure and temperature rises in the pressure suppression system to below design values (Appendix E-II 2.2.3 p. E-78 and the Fifth Supplement).* Each containment spray system is considered operable when both pumps are capable of delivering at least 3000 gpm at a pump developed head of 375 feet of water at 60°F. Requiring both pumps in both systems operable (400 percent redundancy) will assure the availability of the containment spray system.

Allowable outages are specified to account for components that become inoperable in both systems and for more than one component in a system.

The corresponding raw water cooling system is designed to maintain containment spray water temperature no greater than 140°F under the most limiting operating conditions. The containment spray raw water cooling system is considered operable when the flow rate is not less than 3000 gpm and the pressure on the raw water side of the containment spray heat exchangers is not less than 160 psig. The higher pressure on the raw water side will assure that any leakage is into the containment spray system.

Electrical power for all system components is normally available from the reserve transformer. Upon loss of this service the pumping requirement will be supplied from the diesel generator. At least one diesel generator shall always be available to provide backup electrical power for one containment spray system, corresponding raw water cooling system and associated electronic equipment required for automatic system initiation.

Automatic initiation of the containment spray system assures that the containment will not be overpressurized due to hydrogen generation. This automatic feature would only be required if all core spray system malfunctioned and significant metal-water reaction occurred. For the normal condition of 90°F suppression chamber water and 2 psig containment pressure, containment spray actuation would not be necessary for about 15 minutes. Raw water cooling affects the temperature of the spray water and the suppression chamber pool. Taking into

operation

*FSAR

The radioactive liquid and gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid and gaseous effluents during actual or potential releases of liquid and gaseous effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the Offsite Dose Calculation Manual to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring and controlling the concentrations of potentially explosive gas mixtures in the main condenser offgas treatment system. The operability and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50. The purpose of tank level indicating devices, to assure the detection and control of leaks that if not controlled could potentially result in the transport of radioactive materials to unrestricted areas.

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TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION (1) (6)

License	Normal Operation	Shutdown Condition	Operation (3) W/O Process Computer	Reactor Startups
Senior Operator	1	1 (5)	1	1
Operator	2	1	2	3
Unlicensed (2)	2	1	3	2
Asst. Station Shift Supevisor (Shift Technical Advisor Function) (Senior Operator License) (7)	1	1 (4)	1	1

Notes:

(1) At any one time, more licensed or unlicensed operating people could be present for maintenance, repairs, refuel outages, etc.

(2) Those operating personnel not holding an "Operator" or "Senior Operator" License.

(3) For operation longer than eight hours without process computer.

(4) Hot shutdown condition only.

(5) An additional senior reactor operator who has no other concurrent responsibilities shall supervise all core alterations.

(6) The Shift Crew Composition may be one less than the minimum requirements of Table 6.2-1 for a period of time not to exceed two hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the Shift Crew Composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

(7) The Assistant Station Shift Supervisor performs the Shift Technical Advisor function during normal operations or hot shutdown and shall hold a senior reactor operator license. Normally, the Assistant Station Shift Supervisor is a combined Assistant Station Shift Supervisor/Shift Technical Advisor; however, there may be instances when a shift may be staffed by two Senior Reactor Operators plus a dedicated Shift Technical Advisor.