(3) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source, or special nuclear materials as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source, or special nuclear materials without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate such byproduct, source, or special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter 1 and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

AmerGen Energy Company, LLC is authorized to operate the facility at steady-state power levels not in excess of 1930 megawatts (thermal) (100 percent rated power) in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 266, are hereby incorporated in the license. AmerGen Energy Company, LLC shall operate the facility in accordance with the Technical Specifications.

(3) Fire Protection

AmerGen Energy Company, LLC shall implement and maintain in effect all provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report for the facility and as approved in the Safety Evaluation Report dated March 3, 1978, and supplements thereto, subject to the following provision:

* Implemented

1.28 FRACTION OF RATED POWER (FRP)

The FRACTION OF RATED POWER is the ratio of core THERMAL POWER to RATED THERMAL POWER.

1.29 TOP OF ACTIVE FUEL (TAF) - 353.3 inches above vessel zero.

1.30 REPORTABLE EVENT

A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

1.31 IDENTIFIED LEAKAGE

IDENTIFIED LEAKAGE is that leakage which is collected in the primary containment equipment drain tank and eventually transferred to radwaste for processing.

1.32 UNIDENTIFIED LEAKAGE

UNIDENTIFIED LEAKAGE is all measured leakage that is other than identified leakage.

1.33 PROCESS CONTROL PLAN

The PROCESS CONTROL PLAN shall contain the current formulas, sampling, analyses, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61 and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

1.34 AUGMENTED OFFGAS SYSTEM (AOG)

The AUGMENTED OFFGAS SYSTEM is a system designed and installed to holdup and/or process radioactive gases from the main condenser offgas system for the purpose of reducing the radioactive material content of the gases before release to the environs.

1.35 MEMBER OF THE PUBLIC

A MEMBER OF THE PUBLIC is a person who is not occupationally associated with AmerGen Energy Company, LLC and who does not normally frequent the Oyster Creek Nuclear Generating Station site. The category does not include contractors, contractor employees, vendors, or persons who enter the site to make deliveries, to service equipment, work on the site, or for other purposes associated with plant functions.

1.36 OFFSITE DOSE CALCULATION MANUAL (ODCM)

The OFFSITE DOSE CALCULATION MANUAL shall contain the methodology and

1.49 RATED THERMAL POWER (RTP)

RTP shall be a total reactor core heat transfer rate to the reactor coolant of 1930 MWt.

1.50 THERMAL POWER

THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

SECTION 2

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMIT - FUEL CLADDING INTEGRITY

Applicability: Applies to the interrelated variables associated with fuel thermal behavior.

Objective: To establish limits on the important thermal hydraulic variables to assure the integrity of the fuel cladding.

Specifications:

A. When the reactor pressure is greater than or equal to 800 psia and the core flow is greater than or equal to 10% of rated, the existence of a minimum CRITICAL POWER RATIO (MCPR) less than 1.10 for both four or five loop operation and 1.12 for three loop operation shall constitute violation of the fuel cladding integrity safety limit.

B. When the reactor pressure is less than 800 psia or the core flow is less than 10% of rated, the core THERMAL POWER shall not exceed 25% of RATED THERMAL POWER.

C. In the event that reactor parameters exceed the limiting safety system settings in Specification 2.3 and a reactor scram is not initiated by the associated protective instrumentation, the reactor shall be brought to, and remain in, the COLD SHUTDOWN CONDITION until an analysis is performed to determine whether the safety limit established in Specification 2.1.A and 2.1.B was exceeded.

D. During all modes of reactor operation with irradiated fuel in the reactor vessel, the water level shall not be less than 4'8" above the TOP OF ACTIVE FUEL.

2.3 LIMITING SAFETY SYSTEM SETTINGS

Applicability: Applies to trip settings on automatic protective devices related to variables on which safety limits have been placed.

Objective: To provide automatic corrective action to prevent the safety limits from being exceeded.

Specification: Limiting safety system settings shall be as follows:

FUNCTION	LIMITING SAFETY SYSTEM SETTINGS
A. Neutron Flux, Scram	
A.1 APRM	When the reactor mode switch is in the Run position, the APRM flux scram setting shall be the minimum of:

For $W \geq 0.0 \times 10^6$ lb / hr:

$$S \leq \left[\left(0.90 \times 10^{-6}\right) W + 65.1\right] \frac{FRP}{MFLPD} \quad ; \text{or}$$

The applicable stability protection settings, as defined in the COLR,

with a maximum setpoint of 120.0% for core flow equal to 61×10^6 lb/hr and greater,

where:

S = setting in percent of rated power
W = recirculation flow (lb/hr)

FRP = fraction of RATED THERMAL POWER is the ratio of core THERMAL POWER to RATED THERMAL POWER

MFLPD = maximum fraction of limiting power density where the limiting power density for each bundle is the design linear heat generation rate for that bundle.

The ratio of FRP/MFLPD shall be set equal to 1.0 unless the actual operating value is less than 1.0 in which case the actual operating value will be used.

This adjustment may be accomplished by increasing the APRM gain and thus reducing the flow reference APRM High Flux Scram Curve by the reciprocal of the APRM gain change.

A.2 IRM	≤ 38.4 percent of rated neutron flux
A.3 APRM Downscale	≥ 2% RATED THERMAL POWER coincident with IRM Upscale (high-high) or Inoperative

Individual electromatic relief valve control switches shall not be placed in the "Off" position for more than 8 hours (total time for all control switches) in any 30-day period and only one relief valve control switch may be placed in the "Off" position at a time.

i. With two core spray systems OPERABLE:

1. A maximum of two core spray booster pump differential pressure (d/p) switches may be inoperable provided that the switches are in opposing ADS trip system [i.e., only: either RV-40 A&D or RV-40 B&C]. Place the relay contacts associated with the inoperable d/p switch(es) in the de-energized position, within 24 hours. Restore the inoperable d/p switch(es) within 8 days, or declare ADS inoperable and take the action required by Specification 3.4.B.3;

or,

2. If two inoperable d/p switches are in the same ADS trip system [i.e., RV-40 A&B or RV-40 C&D], place the relay contacts associated with the inoperable d/p switch(es) in the de-energized position, within 24 hours. Restore the inoperable d/p switches within 4 days, or declare ADS inoperable and take the action required by Specification 3.4.B.3.

With only one core spray system OPERABLE:

If one or more d/p switches become inoperable in the OPERABLE core spray system, declare ADS inoperable and take the action required by Specification 3.4.B.3.

j. Not required below 40% of rated reactor THERMAL POWER.

k. All four (4) drywell pressure instrument channels may be made inoperable during the integrated primary containment leakage rate test (See Specification 4.5), provided that the plant is in the COLD SHUTDOWN condition and that no work is performed on the reactor or its connected systems which could result in lowering the reactor water level to less than 4'8" above the TOP OF THE ACTIVE FUEL.

l. Bypass in IRM Ranges 8, 9, and 10.

m. There is one time delay relay associated with each of two pumps.

n. One time delay relay per pump must be OPERABLE.

2. The circuit breaker of the recirculation pump motor generator set associated with an ISOLATED RECIRCULATION LOOP shall be open and defeated from operation.

3. An ISOLATED RECIRCULATION LOOP shall not be returned to service unless the reactor is in the COLD SHUTDOWN condition.

b. When there are two inoperable recirculation loops (either two IDLE RECIRCULATION LOOPS or one IDLE RECIRCULATION LOOP and one ISOLATED RECIRCULATION LOOP) the reactor core THERMAL POWER shall not exceed 90% of rated power.

3. If Specifications 3.3.F.1 and 3.3.F.2 are not met, an orderly shutdown shall be initiated immediately until all operable control rods are fully inserted and the reactor is in either the REFUEL MODE or SHUTDOWN CONDITION within 12 hours.

4. With reactor coolant temperature greater than 212°F and irradiated fuel in the reactor vessel, at least one recirculation loop discharge valve and its associated suction valve shall be in the full open position.

5. If Specification 3.3.F.4 is not met, immediately open one recirculation loop discharge valve and its associated suction valve.

6. With reactor coolant temperature less than 212°F and irradiated fuel in the reactor vessel, at least one recirculation loop discharge valve and its associated suction valve shall be in the full open position unless the reactor vessel is flooded to a level above 185 inches TAF or unless the steam separator and dryer are removed.

5. <u>Pressure Suppression Chamber - Drywell Vacuum Breakers</u>

a. When primary containment is required, all suppression chamber-drywell vacuum breakers shall be OPERABLE except during testing and as stated in Specification 3.5.A.5.b and c, below. Suppression chamber - drywell vacuum breakers shall be considered OPERABLE if

(1) The valve is demonstrated to open from closed to fully open with the applied force at all valve positions not exceeding that equivalent to 0.5 psi acting on the suppression chamber face of the valve disk.

(2) The valve disk will close by gravity to within not greater than 0.10 inch of any point on the seal surface of the disk when released after being opened by remote or manual means.

(3) The position alarm system will annunciate in the control room if the valve is open more than 0.10 inch at any point along the seal surface of the disk.

b. Five of the fourteen suppression chamber - drywell vacuum breakers may be inoperable provided that they are secured in the closed position. With one of the nine required suppression chamber-drywell vacuum breakers inoperable, restore one vacuum breaker to OPERABLE status within 72 hours.

c. One position alarm circuit for each OPERABLE vacuum breaker may be inoperable, provided that each OPERABLE suppression chamber – drywell vacuum breaker with one defective alarm circuit, and associated remaining position alarm circuit are verified to be OPERABLE immediately, and monthly in accordance with 4.5.F.5.a. Additionally, a daily verification using the OPERABLE position alarm circuit that the affected vacuum breaker is closed shall be performed.

d. If Specifications3.5.A.5(a), (b) or (c) can not be met, the reactor shall be PLACED IN the COLD SHUTDOWN CONDITION within 24 hours.

6. The primary containment oxygen concentration shall be less than 4.0 volume percent while in the RUN MODE, except as specified in 3.5.A.6.a and 3.5.A.6.b below.

a. From 24 hours after THERMAL POWER is > 15% RTP following startup, to,

b. 24 hours prior to reducing THERMAL POWER to < 15% RTP prior to the next scheduled reactor shutdown.

c. If the primary containment oxygen concentration is greater than or equal to 4.0 volume percent while in the RUN MODE, except as specified in 3.5.A.6.a or 3.5.A.6.b above, restore oxygen concentration to < 4.0 volume percent within 24 hours, otherwise reduce THERMAL POWER to ≤ 15% RTP within the next 8 hours.

7. Deleted.

4.2 REACTIVITY CONTROL

Applicability: Applies to the surveillance requirements for reactivity control.

Objective: To verify the capability for controlling reactivity.

Specification:

A. SDM shall be verified:
   1. Prior to each CORE ALTERATION, and
   2. Once within 4 hours following the first criticality following any CORE ALTERATION.

B. The control rod drive housing support system shall be inspected after reassembly.

C. The maximum scram insertion time of the control rods shall be demonstrated through measurement and, during single control rod scram time tests, the control rod drive pumps shall be isolated from the accumulators:

   1. For all control rods prior to THERMAL POWER exceeding 40% power with reactor coolant pressure greater than 800 psig, following core alterations or after a reactor shutdown that is greater than 120 days.

   2. For specifically affected individual control rods following maintenance on or modification to the control rod or control rod drive system which could affect the scram insertion time of those specific control rods in accordance with either "a" or "b" as follows:

      a.1 Specifically affected individual control rods shall be scram time tested with the reactor depressurized and the scram insertion time from the fully withdrawn position to 90% insertion shall not exceed 2.2 seconds, and

      a.2 Specifically affected individual control rods shall be scram time tested at greater than 800 psig reactor coolant pressure prior to exceeding 40% power.

      b. Specifically affected individual control rods shall be scram time tested at greater than 800 psig reactor coolant pressure.

   3. On a frequency of less than or equal to once per 180 days of cumulative power operation, for at least 20 control rods, on a rotating basis, with reactor coolant pressure greater than 800 psig.

D. Each partially or fully withdrawn control rod shall be exercised at least once each week. This test shall be performed within 24 hours in the event power operation is continuing with two or more inoperable control rods or in the event power operation is continuing with one fully or partially withdrawn rod which cannot be moved and for which control rod drive mechanism damage has not been ruled out. The surveillance need not be completed within 24 hours if the number of inoperable rods has been reduced to less than two and if it has been demonstrated that control rod drive mechanism collet housing failure is not the cause of an immovable control rod.

4.6 RADIOACTIVE EFFLUENT

Applicability: Applies to monitoring of gaseous and liquid radioactive effluents of the Station during release of effluents via the monitored pathway(s). Each Surveillance Requirement applies whenever the corresponding Specification is applicable unless otherwise stated in an individual Surveillance Requirement. Surveillance Requirements do not have to be performed on inoperable equipment.

Objective: To measure radioactive effluents adequately to verify that radioactive effluents are as low as is reasonable achievable and within the limit of 10 CFR Part 20.

Specification:

A. Reactor Coolant

Reactor coolant shall be sampled and analyzed at least once every 72 hours for DOSE EQUIVALENT I-131 during RUN MODE, STARTUP MODE and SHUTDOWN CONDITION.

B. NOT USED.

C. Radioactive Liquid Storage

1. Liquids contained in the following tanks shall be sampled and analyzed for radioactivity at least once per 7 days when radioactive liquid is being added to the tank:

   a. Waste Surge Tank, HP-T-3;
   b. Condensate Storage Tank.

D. Main Condenser Offgas Treatment

RELOCATED TO THE ODCM.

E. Main Condenser Offgas Radioactivity

1. The gross radioactivity in fission gases discharged from the main condenser air ejector shall be measured by sampling and analyzing the gases.

   a. at least once per month, and

   b. When the reactor is operating at more than 40 percent of rated power, within 4 hours after an increase in the fission gas release via the air ejector of more than 50 percent, as indicated by the Condenser Air Ejector Offgas Radioactivity Monitor after factoring out increase(s) due to change(s) in the THERMAL POWER level.

F. Condenser Offgas Hydrogen Concentration

The concentration of hydrogen in offgases downstream of the recombiner in the Offgas System shall be monitored with hydrogen instrumentation as described in Table 3.15.2.

G. NOT USED.

H. NOT USED.