October 31, 1986

Docket No. 50-219

Mr. P. B. Fiedler
Vice President and Director
Oyster Creek Nuclear Generating Station
Post Office Box 388
Forked River, New Jersey 08731

Dear Mr. Fiedler:

SUBJECT: HIGH DRYWELL PRESSURE SETPOINT AND ISOLATION CONDENSER HIGH FLOW TRIP BYPASS (TSCR 147, TAC 62977 AND 63015)

Re: Oyster Creek Nuclear Generating Station

The Commission has issued the enclosed Amendment No. 112 to Provisional Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station. This amendment is in response to your application dated September 11, 1986.

This amendment authorizes changes to the Appendix A Technical Specifications (TS). These changes (1) increase the high drywell pressure trip setpoint limit to 3.5 psig, (2) add a bypass to the high flow trip of the "B" Isolation Condenser when initiating alternate shutdown and (3) revise the appropriate Bases of the TS. These are changes to Table 3.1.1, Protective Instrumentation Requirements, and to the Bases of Section 3.1, Protective Instrumentation, of the TS.

The State of New Jersey has expressed concerns about your request to add a bypass to the high flow trip of the "B" Isolation Condenser. These are addressed in Section 3.4 of the enclosed Safety Evaluation for this license amendment.

A copy of our related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notices.

Sincerely,

Jack N. Donohew, Jr., Project Manager
BWR Project Directorate #1
Division of BWR Licensing

10/30/86 10/30/86

Enclosures:
1. Amendment No. 112 to License No. DPR-16
2. Safety Evaluation

cc w/enclosures:
See next page

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Mr. P. B. Fiedler
Oyster Creek Nuclear Generating Station

Oyster Creek Nuclear
Generating Station

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

GPU NUCLEAR CORPORATION

AND

JERSEY CENTRAL POWER & LIGHT COMPANY

DOCKET NO. 50-219

OYSTER CREEK NUCLEAR GENERATING STATION

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 112
License No. DPR-16

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by GPU Nuclear Corporation and Jersey Central Power and Light Company (the licensees) dated September 11, 1986, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of Provisional Operating License No. DPR-16 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 112, are hereby incorporated in the license. GPU Nuclear Corporation shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Jack N. Donohew, Jr.
Jack N. Donohew, Jr., Project Manager
BWR Project Directorate #1
Division of BWR Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 31, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 112

PROVISIONAL OPERATING LICENSE NO. DPR-16

DOCKET NO. 50-219

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by the captioned amendment number and contain vertical lines indicating the area of change.

REMOVE	INSERT
3.1-4	3.1-4
3.1-5	3.1-5
3.1-6	3.1-6
3.1-7	3.1-7
3.1.8	3.1.8
3.1-11	3.1-11
3.1-12	3.1-12
3.1-13	3.1-13
3.1-17	3.1-17

particular protection instrument is not required; or the plant is placed in the protection or safe condition that the instrument initiates. This is accomplished in a normal manner without subjecting the plant to abnormal operating conditions. The action and out-of-service requirements apply to all instrumentation within a particular function, e.g., if the requirements on any one of the ten scram functions cannot be met then control rods shall be inserted.

The trip level settings not specified in Specification 2.3 have been included in this specification. The bases for these settings are discussed below.

The high drywell pressure trip setting is $\leq$3.5 psig. This trip will scram the reactor, initiate reactor isolation, initiate containment spray in conjunction with low low reactor water level, initiate core spray, initiate primary containment isolation, initiate automatic depressurization in conjunction with low-low-low-reactor water level, initiate the standby gas treatment system and isolate the reactor building. The scram function shuts the core down during the loss-of-coolant accidents. A steam leak of about 15 gpm and a liquid leak of about 35 gpm from the primary system will cause drywell pressure to reach the scram point; and, therefore the scram provides protection for breaks greater than the above.

High drywell pressure provides a second means of initiating the core spray to mitigate the consequences of a loss-of-coolant accident. Its trip setting of $\leq$3.5 psig initiates the core spray in time to provide adequate core cooling. The break-size coverage of high drywell pressure was discussed above. Low-low water level and high drywell pressure in addition to initiating core spray also causes isolation valve closure. These settings are adequate to cause isolation to minimize the offsite dose within required limits.

It is permissible to make the drywell pressure instrument channels inoperable during performance of the integrated primary containment leakage rate test provided the reactor is in the cold shutdown condition. The reason for this is that the Engineered Safety Features, which are effective in case of a LOCA under these conditions, will still be effective because they will be activated by low-low reactor water level.

The scram discharge volume has two separate instrument volumes utilized to detect water accumulation. The high water level is based on the design that the water in the SDIV's, as detected by either set of level instruments, shall not be allowed to exceed 29.0 gallons; thereby, permitting 137 control rods to scram. To provide further margin, an accumulation of not more than 14.0 gallons of water, as detected by either instrument volume, will result in a rod block and an alarm. The accumulation of not more than 7.0 gallons of water, as detected in either instrument volume will result in an alarm.

Detailed analyses of transients have shown that sufficient protection is provided by other scrams below 45% power to permit bypassing of the turbine trip and generator load rejection scrams. However, for operational convenience, 40% of rated power has been chosen as the setpoint below which these trips are bypassed. This setpoint is coincident with bypass valve capacity.

A low condenser vacuum scram trip of 23" Hg has been provided to protect the main condenser in the event that vacuum is lost. A loss of condenser vacuum would cause the turbine stop valves to close, resulting in a turbine trip transient. The low condenser vacuum trip anticipates this transient and scrams the reactor. The condenser is capable of receiving bypass steam until 7" Hg vacuum thereby mitigating the transient and providing a margin.

Main steamline high radiation is an indication of excessive fuel failure. Scram and reactor isolation are initated when high activity is detected in the main steam lines. These actions prevent further release of fission products to the environment. This is accomplished by setting the trip at 10 times normal rated power background. Although these actions are initiated at this level, at lower activities the monitoring system also provides for continuous monitoring of radioactivity in the primary steam lines as discussed in Section VII-6 of the FDSAR. Such capability provides the operator with a prompt indication of any release of fission products from the fuel to the reactor coolant above normal rated power background. The gross failure of any single fuel rod could release a sufficient amount of activity to approximately double the background activity at normal rated power. This would be indicative of the onset of fuel failures and would alert the operator to the need for appropriate action, as defined by Section 6 of these specifications.

The settings to isolate the isolation condenser in the event of a break in the steam or condensate lines are based on the predicted maximum flows that these systems would experience during operation, thus permitting operation while affording protection in the event of a break. The settings correspond to a flow rate of less than three times the normal flow rate of 3.2×10^5 lb/hr. Upon initiation of the alternate shutdown panel, this function is bypassed to prevent spurious isolation due to fire induced circuit faults.

The setting of ten times the stack release limit for isolation of the air-ejector offgas line is to permit the operator to perform normal, immediate remedial action if the stack limit is exceeded. The time necessary for this action would be extremely short when considering the annual averaging which is allowed under 10 CFR 20.106, and, therefore, would produce insignificant effects on doses to the public.

Four radiation monitors are provided which initiate isolation of the reactor building and operation of the standby gas treatment system. Two monitors are located in the ventilation ducts, one is located in the area of the refueling pool and one is located in the reactor vessel head storage area. The trip logic is basically a 1 out of 4 system. Any upscale trip will cause the desired action. Trip settings of 17 mr/hr in the duct and 100 mr/hr on the refueling floor are based upon initiating standby gas treatment system so as not to exceed allowed dose rates of 10 CFR 20 at the nearest site boundary.

The SRM upscale of 5×10^5 CPS initiates a rod block so that the chamber can be relocated to a lower flux area to maintain SRM capability as power is increased to the IRM range. Full scale reading is 1×10^6 CPS. This rod block is bypassed in IRM Ranges 8 and higher since a level of 5×10^5 CPS is reached and the SRM chamber is at its fully withdrawn position.

The SRM downscale rod block of 100 CPS prevents the instrument chamber from being withdrawn too far from the core during the period that it is required to monitor the neutron flux. This downscale rod block is also bypassed in IRM

Ranges 8 and higher. It is not required at this power level since good indication exists in the Intermediate Range and the SRM will be reading approximately 5×10^5 CPS when using IRM Ranges 8 and higher.

The IRM downscale rod block in conjunction with the chamber full-in position and range switch setting, provides a rod block to assure that the IRM is in its most sensitive condition before startup. If the two latter conditions are satisfied, control rod withdrawal may commence even if the IRM is not reading at least 5%. However, after a substantial neutron flux is obtained, the rod block setting prevents the chamber from being withdrawn to an insensitive area of the core.

The APRM downscale setting of $\geq$ 2/150 full scale is provided in the run mode to prevent control rod withdrawal without adequate neutron monitoring.

High flow in the main steamline is set at 120% of rated flow. At this setting the isolation valves close and in the event of a steam line break limit the loss of inventory so that fuel clad perforation does not occur. The 120% flow would correspond to the thermal power so this would either indicate a line break or too high a power.

Temperature sensors are provided in the steam line tunnel to provide for closure of the main steamline isolation valves should a break or leak occur in this area of the plant. The trip is set at 50°F above ambient temperature at rated power. This setting will cause isolation to occur for main steamline breaks which result in a flow of a few pounds per minute or greater. Isolation occurs soon enough to meet the criterion of no clad perforation.

The low-low-low water level trip point is set at 4'8" above the top of the active fuel and will prevent spurious operation of the automatic relief system. The trip point established will initiate the automatic depressurization system in time to provide adequate core cooling.

Specification 3.1.B.1 defines the minimum number of APRM channel inputs required to permit accurate average core power monitoring. Specifications 3.1.B.2 and 3.1.C.1 further define the distribution of the operable chambers to provide monitoring of local power changes that might be caused by a single rod withdrawal. Any nearby, operable LPRM chamber can provide the required input for average core monitoring. A Travelling Incore Probe or Probes can be used temporarily to provide APRM input(s) until LPRM replacement is possible. Since APRM rod block protection is not required below 61% of rated power,(1) as discussed in Section 2.3, Limiting Safety System Settings, operation may continue below 61% as long as Specification 3.1.B.1 and the requirements of Table 3.1.1 are met. In order to maintain reliability of core monitoring in that quadrant where an APRM is inoperable, it is permitted to remove the operable APRM from service for calibration and/or test provided that the same core protection is maintained by alternate means.

In the rare event that Travelling In-core Probes (TIPs) are used to meet the requirements 3.1.B or 3.1.C, the licensee may perform an analysis of substitute LPRM inputs to the APRM system using spare (non-APRM input) LPRM detectors and change the APRM system as permitted by 10 CFR 50.59.

Under assumed loss-of-coolant accident conditions and under certain loss of offsite power conditions with no assumed loss-of-coolant accident, it is inadvisable to allow the simultaneous starting of emergency core cooling and heavy load auxiliary systems in order to minimize the voltage drop across the emergency buses and to protect against a potential diesel generator overload. The diesel generator load sequence time delay relays provide this protective function and are set accordingly. The repetitive accuracy rating of the timer mechanism as well as parametric analyses to evaluate the maximum acceptable tolerances for the diesel loading sequence timers were considered in the establishment of the appropriate load sequencing.

Manual actuation can be accomplished by the operator and is considered appropriate only when the automatic load sequencing has been completed. This will prevent simultaneous starting of heavy load auxiliary systems and protect against the potential for diesel generator overload.

Also, the Closed Cooling Water and Service Water pump circuit breakers will trip whenever a loss-of-coolant accident condition exists. This is justified by Amendment 42 of the Licensing Application which determined that these pumps were not required during this accident condition.

Reference:

(1) NEDO-10189 "An Analysis of Functional Common Mode Failures in GE BWR Protection and Control Instrumentation," L. G. Frederick, et al., July 1970.

TABLE 3.1.1 PROTECTIVE INSTRUMENTATION REQUIREMENTS

Function	Trip Setting	Reactor Modes in which Function Must Be Operable: Shutdown	Refuel	Startup	Run	Min. No. of Operable or Operating [tripped] Trip Systems	Min. No. of Instrument Channels Per Operable Trip Systems	Action Required*
A. Scram								Insert control rods
1. Manual Scram		X	X	X	X	2	1	
2. High Reactor Pressure	**		X(s)	X	X	2	2	
3. High Drywell Pressure	≤ 3.5 psig.		X(u)	X(u)	X	2	2	
4. Low Reactor Water Level	**		X	X	X	2	2	
5. a. High Water Level in Scram Discharge Volume North Side	≤ 29 gal.		X(a)	X(z)	X(z)	2	2	
b. Higher Water Level in Scram Discharge Volume South Side	≤ 29 gal.		X(a)	X(z)	X(z)	2	2	
6. Low Condenser Vacuum	≥ 23" hg.		X(b)	X(b)	X	2	2	
7. High Radiation in Main Steam Line Tunnel	≤ 10 x normal background		X(s)	X	X	2	2	

TABLE 3.1.1 PROTECTIVE INSTRUMENTATION REQUIREMENTS (CONTD)

Function	Trip Setting	Reactor Modes in which Function Must Be Operable: Shutdown	Refuel	Startup	Run	Min. No. of Operable or Operating [tripped] Trip Systems	Min. No. of Instrument Channels Per Operable Trip Systems	Action Required*
D. Core Spray								Consider the respective core spray loop inoperable, and comply with Spec. 3.4
1. Low-Low Reactor Water Level	**	X(t)	X(t)	X(t)	X	2	2	
2. High Drywell Pressure	≤ 3.5 psig	X(t)	X(t)	X(t)	X	2(k)	2(k)	
3. Low Reactor Pressure (valve permissive)	≥ 285 psig	X(t)	X(t)	X(t)	X	2	2	
E. Containment Spray								Consider the containment spray loop inoperable and comply with Spec. 3.4
1. High Drywell Pressure	≤ 3.5 psig	X(u)	X(u)	X(u)	X	2(k)	2(k)	
2. Low-Low Reactor Water Level	≥ 7'2" above top of active fuel	X(u)	X(u)	X(u)	X	2	2	
F. Primary Containment Isolation								Isolate containment or place in cold shutdown condition
1. High Drywell Pressure	≤ 3.5 psig	X(u)	X(u)	X(u)	X	2(k)	2(k)	
2. Low-Low Reactor Water Level	≥ 7'2" above top of active fuel	X(u)	X(u)	X(u)	X	2	2	

TABLE 3.1.1 PROTECTIVE INSTRUMENTATION REQUIREMENTS (CONTD)

Function	Trip Setting	Reactor Modes in which Function Must Be Operable: Shutdown	Refuel	Startup	Run	Min. No. of Operable or Operating [tripped] Trip Systems	Min. No. of Instrument Channels Per Operable Trip Systems	Action Required*
G. Automatic Depressurization								
1. High Drywell Pressure	$\leq$ 3.5 psig	X(v)	X(v)	X(v)	X	2(k)	2(k)	See note h
2. Low-Low-Low Reactor Water Level	$\geq$ 4'8" above top of active fuel	X(v)	X(v)	X(v)	X	2	2	See note(
3. AC Voltage	NA			X(v)	X	2	2	Prevent auto depressurization on loss of AC power. See note i
H. Isolation Condenser Isolation (See Note hh)								
1. High Flow Steam Line	$\leq$ 20 psig P	X(s)	X(s)	X	X	2	2	Isolate Affected Isolation condensor, comply with Spec. 3.8 See note (
2. High Flow Condensate Line	$\leq$ 27" P H_2O	X(s)	X(s)	X	X	2	2	
I. Offgas System Isolation								
1. High Radiation In Offgas Line (e)	$\leq$ 10 x Stack Release limit (See 3.6-A.1)	X(s)	X(s)	X	X	1	2	Isolate reactor or trip the Inoperable instrument channel

TABLE 3.1.1 PROTECTIVE INSTRUMENTATION REQUIREMENTS (CONTD)

Function	Trip Setting	Reactor Modes in which Function Must Be Operable: Shutdown	Refuel	Startup	Run	Min. No. of Operable or Operating [tripped] Trip Systems	Min. No. of Instrument Channels Per Operable Trip Systems	Action Required*
J. Reactor Building Isolation and Standby Gas Treatment System Initiation								
1. High Radiation Reactor Building Operation Floor	≤ 100 Mr/Hr	X(w)	X(w)		X	1	1	Isolate Reactor Bldg. and Initiate Standby Gas Treatment System or Manual Surveillance for not more than 24 hours
2. Reactor Bldg. Ventilation Exhaust	≤ 17 Mr/Hr	X(w)	X(w)	X	X	1	1	
3. High Drywell Pressure	≤ 3.5 psig	X(u)	X(u)	X	X	1(k)	2(k)	(total for all instruments under J) in any 30-day period
4. Low Low Reactor Water Level	≥ 7'2" above top of active fuel	X(gg)	X	X	X	1	2	
K. Rod Block								No control rod withdrawals permitted
1. SRM Upscale	≤ 5×10^5 cps		X	X(1)		1	2	
2. SRM Downscale	≥ 100 cps(f)		X	X(1)		1	2	
3. IRM Downscale	≥ 5/125 fullscale(g)		X	X		2	3	
4. APRM Upscale	**		X(s)	X	X	2	3(c)	
5. APRM Downscale	≥ 2/150 fullscale				X	2	3(c)	

TABLE 3.1.1 (CONTD)

*gg. These functions are not required to be operable when secondary containment is not required to be maintained or when the conditions of Sections 3.5.b.1.a, b, c, and d are met, and reactor water level is closely monitored and logged hourly. The Standby Gas Treatment System will be manually initiated if reactor water level drops to the low level trip set point.

hh. The high flow trip function for "B" Isolation Condenser is bypassed upon initiation of the alternate shutdown panel. This prevents a spurious trip of the isolation condenser in the event of fire induced circuit damage.

* This note is applicable only during the Cycle 10M outage.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 112 TO PROVISIONAL OPERATING LICENSE NO. DPR-16

GPU NUCLEAR CORPORATION AND

JERSEY CENTRAL POWER & LIGHT COMPANY

OYSTER CREEK NUCLEAR GENERATING STATION

DOCKET NO. 50-219

1.0 INTRODUCTION

By letter dated September 11, 1986, GPU Nuclear (the licensee) requested an amendment to Provisional Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station (Oyster Creek). This amendment would authorize changes to the Appendix A Technical Specifications (TS). These changes would (1) increase the high drywell pressure trip setpoint limit to 3.5 psig, (2) add a bypass to the high flow trip to the "B" Isolation Condenser when initiating the alternate shutdown and (3) revise the appropriate Bases of the TS. These are changes to Table 3.1.1, Protective Instrumentation Requirements, and the Bases of Section 3.1, Protective Instrumentation, of the TS.

2.0 DISCUSSION AND EVALUATION

The proposed amendment would (1) increase the high drywell pressure trip setpoint from not greater than 2.4 psig to not greater than 3.5 psig for the instrumentation to actuate its associated trip systems and (2) add a bypass to the high steam line flow trip and high condensate return line flow trip for isolation of the "B" Isolation Condenser for when the alternate shutdown panel is initiated. The associated trip systems for the high drywell pressure instrumentation cause reactor scram, core spray initiation, containment spray initiation, containment isolation, automatic reactor vessel depressurization and reactor building isolation and standby gas treatment system (SGTS) initiation. For the bypass, the licensee is proposing to add a footnote "hh" stating that the trip function is bypassed upon initiation of the alternate shutdown panel to prevent a spurious trip of the "B" Isolation Condenser in the event of fire induced circuit damage.

The instrument setpoint for the High Drywell Pressure TS limit of 2.4 psig was found by the licensee to be unacceptable to maintain and achieve safe shutdown conditions during a postulated Appendix R event. This event involves a fire. For this event, the drywell cooling fans are assumed to be lost due to the fire and the reactor is being cooled by the Isolation Condenser with no feedwater flow. Thus, it is essential that the Automatic Depressurization System (ADS) logic is not actuated by high drywell pressure to further reduce reactor water level.

The licensee concluded in its analysis for this event that the drywell pressure could exceed the current high drywell pressure instrument setpoint when you take into account the required adjustments to the instrumentation to have its setpoint less than the existing 2.4 psig TS limit. The actual setpoint incorporated in the high drywell pressure instrumentation setpoint is lower than the TS limit to account for possible instrument error and drift. Therefore, the three ADS actuation logic signals (low-low-low reactor water level, high drywell pressure, and core spray pump discharge pressure) may be satisfied with the existing TS instrument setpoint of 2.4 psig for high drywell pressure and, therefore, result in an inadvertent initiation of ADS. To prevent such an inadvertent actuation of ADS during a postulated Appendix R event and to minimize spurious trips caused by instrument drift, a revised TS limit of 3.5 psig has been requested by the licensee for high drywell pressure.

3.0 EVALUATION

3.1 High Drywell Pressure Setpoint

The licensee has evaluated the acceptability of increasing the drywell pressure TS limit to 3.5 psig and determined the effect of the increased TS limit on anticipated plant operational occurrences and design basis accidents. Each of the protective functions listed below was examined by the licensee to determine how each function would be altered by the new TS limit and how this altered protective function response would affect the plant design response to the transients and accidents evaluated on the Oyster Creek docket. The following are the conclusions by the licensee:

Reactor Scram

The high drywell pressure scram function is provided to shut down the core following a loss-of-coolant accident (LOCA). For most LOCA events, this function will precede a low reactor water level scram signal. However, the Oyster Creek LOCA analyses which were submitted by the licensee in response to 10 CFR 50.46 and 10 CFR Part 50, Appendix K, demonstrate for large breaks that shutdown will occur as a result of excessive voiding, not high drywell pressure. For small breaks, scram will occur at 0.3 second as a result of the loss of offsite power. Therefore, the scrams which would have been associated with high drywell pressure were not the determinant scrams.

Small breaks without a loss of offsite power would be less severe due to feedwater availability. The core is adequately protected by a scram on low reactor water level and a scram associated with main steam isolation valve (MSIV) closure at low-low reactor water level. In any case, the scram delay associated with a drywell pressure TS limit increase to 3.5 psig is minimal and is more than compensated for by the conservative scram reactivity curves used in the analyses. Further, the normal operating drywell pressure is typically greater than atmospheric pressure which is assumed in the analyses, and thus the actual pressure difference and associated time delay is less than in the analysis.

For large steam region breaks with feedwater available, the scram associated with high main steam line flow, low system pressure and low reactor water level would occur at approximately the same time as the high drywell pressure. In these cases, the effect of a TS limit increase to 3.5 psig would be negligible on the transient behavior. For small steam line breaks with feedwater, high drywell pressure is the only scram function. In these cases, the small increases in the setpoint would have a negligible effect on the transient severity. In these cases, the vessel pressure and level remain within normal bounds so that the core is cooled in the normal manner. In this way, even a large delay in scram time on high drywell pressure would not have any impact on this LOCA. The operator could, in fact, proceed with an orderly shutdown if the scram does not occur.

Core Spray Pump Start

The core spray pumps will automatically start on either low-low reactor water level or high drywell pressure. Depending on the nature of the LOCA, either one or both of these signals will be available. In all cases analyzed for Oyster Creek, the time required to depressurize the system to the 285 psig core spray permissive pressure is limiting with respect to core spray initiation. Thus, the core spray flow will begin at the same time for either a 2.4 or 3.5 psig high drywell pressure TS limit.

Containment Spray

The containment spray system will actuate automatically upon indication of high drywell pressure and low-low reactor water level. Depending upon the size and location of the break and whether or not feedwater is available, the high drywell pressure signal will occur either alone or in conjunction with low-low reactor water level. For all break sizes either above or below the core, without feedwater, high drywell pressure will occur prior to low-low reactor water level. A high drywell pressure TS limit increase from 2.4 to 3.5 psig will not change this result. For large breaks with feedwater, this conclusion is also valid. For small breaks with feedwater, low-low reactor water level may not occur and operator action will be required to initiate the sprays. Again, the increased high drywell pressure setpoints will not affect this conclusion. There are no LOCA events analyzed on the docket for which the increase of 1.1 psig in the TS limit will prevent or delay the automatic initiation of the containment spray system.

Primary Containment and Reactor Building Isolation

Primary and secondary containment isolation results automatically from high drywell pressure or low-low reactor water level. The arguments presented above regarding the negligible delay in high drywell pressure indication associated with a 1.1 psig TS limit increase are applicable here. Coupling this with the fact that high drywell pressure precedes low-low reactor water level for all break sizes and locations provides assurance that fuel damage will not have occurred as a result of a LOCA prior to isolation of the containment. The TS limit increase will not alter the order of signal initiation for all breaks analyzed. As indicated previously, even for very small steam breaks, the time delay associated with the 1.1 psig TS limit increase is negligible (approximately 40 seconds of a 0.01 ft^2 main steam line break).

Automatic Depressurization System (ADS)

The actuation of the ADS, which is required during a small break LOCA to depressurize the vessel and permit low pressure core spray flow, is limited in its initiation by the time required to reach low-low-low reactor water vessel level. For small breaks with or without feedwater flow, high drywell pressure will be reached within seconds even for the smallest break analyzed on the Oyster Creek docket. The time required to reach low-low-low reactor water level for these cases is much longer. If feedwater is available, low-low-low reactor water level may not be reached in some cases and will be delayed in all cases. Thus, a high drywell pressure TS limit increase of 1.1 psig will not result in a change in the initiation time of ADS for any small break analyzed.

Standby Gas Treatment System (SGTS) Initiation

The SGTS treats and exhausts the atmosphere of the reactor building to the stack during containment isolation conditions. This prevents ground level leakage of fission products from the reactor building. This system is initiated by high drywell pressure or low-low reactor water level analogous to primary and secondary containment isolation.

The arguments pertaining to reactor building isolation are all applicable to the SGTS. Because both are initiated by the same signals, the SGTS will be available to perform its intended function simultaneously with isolation of the reactor building which is its mode of operation in response to an accident.

Time Delay to Reach 3.5 psig Drywell Pressure

This change results in the initiation of automatic protection actions to protect the drywell at a later time in an accident involving loss of coolant and an increase of pressure in the drywell. As described above, the setpoint is still low enough that the trips initiated by the affected instruments will occur in time to ensure that they will perform their protective function. Based on the design basis LOCA (a large break) inside the drywell in Section XIII-2.4 of the Oyster Creek Unit No. 1 Facility Description and Safety Analysis Report, the new setpoint will initiate trips less than 0.1 second later in time. The analysis in Section XIII-2.4 also ignores the trip that occurs at high drywell pressure because it is based on the initiation of automatic protective functions with the later trip at low-low reactor water level in the core. Therefore, this analysis does not change with this proposed action to raise the trip setpoint for high drywell pressure from 2.4 to 3.5 psig.

3.2 Bypass For the "B" Isolation Condenser Isolation Trip

The following is related to the proposed bypass for "B" Isolation Condenser isolation. There is no bypass for the "A" Isolation Condenser. The alternate shutdown capability is being provided to assure safe shutdown and cooldown of the reactor in the event of a fire causing evacuation of the control room or the loss of control room function due to fire damage in the cable spread rooms. The design of the alternate shutdown system includes bypassing the high flow trip function for high flow in the steam line to and condensate return line from the "B" Isolation Condenser. These trips are to isolate the condenser for a break in either line when the condenser should be isolated to prevent loss of water from the reactor coolant system.

The design of the alternate shutdown system including this bypass was reviewed and approved by the staff in Sections 8.1.4 and 8.2.4.3 of its Safety Evaluation (SE) dated March 24, 1986, based on the licensee's submittals on its Fire Protection Plan dated April 3, July 12 and October 9, 1985. This capability utilizes the isolation condenser for decay heat removal and reactor cooldown to establish a safe shutdown condition. Since a fire affecting cabling associated with the high flow isolation condenser trip function could result in a spurious isolation of the "B" Isolation Condenser, the design includes a bypass of the trip function upon initiation of the alternate shutdown panel.

In Sections 8.1.4 and 8.2.4.3 of its SE, the staff stated that the control logic circuits of the isolation condenser valves will be modified and cables rerouted to allow control of the isolation condenser from the remote shutdown panel, that the devices whose inadvertent operation by spurious signals could affect safe shutdown have been identified as shutdown circuits and are included in the separation analysis and that the licensee will provide isolation and transfer switches for all shutdown circuits as needed to prevent spurious operation. This SE is based on Table D2, page D-22 and on the section, Decay Heat Removal, in Appendix A which discusses the use of the Isolation Condensers to provide hot shutdown capability during a fire. These are in the licensee's Fire Protection Plan. The Table D2 lists the isolation valves V-14-32, V-14-33, V-14-35 and V-14-37, main steam line and condensate return line of the "B" Isolation Condenser, respectively, as subject to possible spurious closure which must be prevented. In Appendix A, the licensee states that the control of the cooldown using the Isolation Condensers requires the operator to keep the steam line valves and one of the condensate return valves open and cycle the other condensate return valve open and close.

The occurrence of a fire which requires initiation of the alternate shutdown panel in conjunction with a line break accident is not considered a credible event. The bypass becomes effective only when the alternate shutdown capability is initiated and this is done by operators at the alternate shutdown panel. The panel is initiated at the panel through transfer switches which are key locked. The keys are locked away in the control room. This is to prevent inadvertent actuation of the panel and inadvertent initiation of the bypass. There is also an alarm in the control room when the panel is actuated to inform the operators that the panel has been actuated. In addition, a single failure of the switch will not preclude operation of the Isolation Condenser high flow trip in the event of a line break accident.

If the Isolation Condensers should not be available or the isolation valves are tripped, the alternate hot shutdown capability in the licensee's Fire Protection Plan is the Electromatic Relief Valve Path. The operators at the alternate shutdown panel have parameters before them as reactor coolant system water level and pressure to indicate the shutdown capability of the Isolation Condensers. The operators can isolate the Isolation Condensers and switch to the Electromatic Relief Valve Path if this is needed.

3.3 Conclusions

The staff has reviewed the licensee's justification in Section 3.1 for increasing the high drywell pressure setpoint to less than or equal to 3.5 psig in Table 3.1.1 of the TS. The staff concludes that this increase will significantly reduce the possibility of unneeded ADS actuation during an event while not changing the accident analysis based on this setpoint. Therefore, the staff concludes that this change and the changes to the Bases describing this change are acceptable.

The staff has also reviewed the justification in Section 3.2 for the high flow trip bypass. The staff has accepted the design of the alternate shutdown system including bypassing the high flow trip function for the "B" Isolation Condenser in its SE dated March 24, 1986. Using this bypass prevents a possible spurious isolation of the "B" Isolation Condenser during a fire. The transfer switches for alternate shutdown are keylocked to prevent inadvertent actuation of the alternate shutdown panel and bypass and no single failure of the transfer switch will preclude operation of the Isolation Condenser high flow trip. Therefore, the staff concludes that this change is acceptable.

3.4 Consultation With The State

On October 24, 1986, the staff's Project Manager consulted with Ms. R. Green and Mr. K. Tosch of the State of New Jersey, Bureau of Radiation Protection. This consultation, in accordance with 10 CFR 50.91(b)(4), was concerned with the staff's intent to issue these TS changes. The State stated that it did not have a concern with the TS change to increase the high drywell pressure setpoint limit but it did have a concern with the bypass for the "B" Isolation Condenser high flow trip.

The concerns of the State were that the Isolation Condenser piping, to and from the reactor vessel, is susceptible to intergranular stress corrosion cracking (IGSCC), has suffered some cracking in the past and is partly located outside the drywell where leaks from these pipes, from small breaks which may not be noticed by the operators, could cause severe environmental consequences offsite. In addition, the isolation valves on the steam lines from the reactor vessel to the Isolation Condensers are both outside the drywell. The State expressed the concern that the staff was not being sufficiently conservative in accepting this bypass in the proposed TS change instead of having the licensee reroute or fix the cable to prevent the chance of a spurious signal in a fire.

The high flow trip function is provided to isolate the system in the event of a line break. The occurrence of a major fire requiring the evacuation of the control room to the alternate shutdown panel and a line break accident is not required by the staff's implementation of Appendix R and is not considered sufficiently credible to be designed for. The Isolation Condenser system was inspected for IGSCC indications by the licensee in the current Cycle 11R outage and the results were submitted to the staff in the licensee's letter dated October 3, 1986. The licensee concluded, on the 18 structural weld overlays on the Isolation Condenser System, that there were no IGSCC indications and the overlays are acceptable for continued operation. The staff will complete its evaluation of these inspections as a separation action, before the plant is restarted from the current outage.

The concern that both containment isolation valves on the steam lines to the Isolation Condensers are on the outside of the drywell is Topic III-5.B, Pipe Break Outside Containment, of the staff's Systematic Evaluation Program (SEP). This is also a separate review by the staff for which the licensee has provided fracture mechanics analyses that the licensee has stated demonstrates that through-wall cracks in the Isolation Condenser steam pipe would open up, yet remain stable, under severe pipe pressure loading and rotation stresses. The analysis concludes that no instantaneous pipe break would occur and the estimated pipe leakage for these cracks would be less than 1 gpm. These lines are in the licensee's inservice inspection program and the inspections are in accordance with Section XI of the ASME Code. The lines are considered adequately sound for continued plant operation until the SEP Topic is resolved.

The proposed TS change on the bypass for the "B" Isolation Condenser high flow trip does not affect how the Isolation Condensers may be used. Therefore, it is not related to the question of cracking in the Isolation Condenser system; however, this system has been inspected during the current outage and the licensee stated it found no IGSCC indications in the weld overlays inspected. Therefore, subject to the staff's separate action on this inspection, the system is considered adequately sound.

Appendix R only requires the evaluation of a loss of offsite power concurrent with the fire and does not require that other unlikely events such as pipe breaks be considered. Therefore, the staff did not require the licensee to reroute or replace the applicable cable to prevent the spurious signal.

However, if a line break should occur after initiation of the alternate shutdown panel, the bypass will affect only the "B" Isolation Condenser. The reactor will already have been scrammed before the break and, as a minimum, the radiation monitors in the stack for the Reactor Building ventilation will isolate the Reactor Building and start up the Standby Gas Treatment System (SGTS) to filter the release from the building to the environment. The release will be within acceptable limits and the operators will have sufficient information to isolate the Emergency Condensers if the high flow trip did not and use the Automatic Depressurization System (ADS) to achieve and maintain hot shutdown. The licensee's Fire Protection Plan dated August 25, 1986, has the Isolation Condensers and the ADS to achieve and maintain hot shutdown.

4.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement nor environmental assessment need be prepared in connection with the issuance of this amendment.

5.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security nor to the health and safety of the public.

6.0 REFERENCES

1. Letter from J. A. Zwolinski (NRC to P. B. Fiedler (GPUN), Exemptions from Requirements of Appendix R to 10 CFR Part 50, Section III.G.2 and the Post Fire Shutdown Capability, dated March 24, 1982.

2. Letter from P. B. Fiedler (GPUN) to J. A. Zwolinski (NRC), TSCR No. 147, dated September 11, 1986.

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