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UNITED STATES NUCLEAR REGULATORY COMMISSION

GPU NUCLEAR CORPORATION

JERSEY CENTRAL POWER AND LIGHT COMPANY

DOCKET NO. 50-219

NOTICE OF CONSIDERATION OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSE AND PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION AND OPPORTUNITY FOR HEARING

The U.S. Nuclear Regulatory Commission (the Commission) is considering issuance of an amendment to Provisional Operating License No. DPR-16 issued to GPU Nuclear Corporation and Jersey Central Power and Light Company, for operation of the Oyster Creek Nuclear Generating Station, located in Ocean County, New Jersey.

The first proposed amendment would revise the footnote marked with an asterisk "*" to Table 3.1.1, Protective Instrumentation Requirements, of the Appendix A Technical Specifications (TS). When it is necessary to conduct tests and calibrations of the protective instrumentative channels in accordance with the TS, the licensee proposes that one channel may be made inoperable for up to 2 hours without tripping the channel's trip system. This is instead of the existing requirement which allows that channel to be inoperable without tripping the trip system for only up to 1 hour per month. This first amendment is in accordance with the licensee's application dated September 5, 1986, for Technical Specification Change Request (TSCR) No. 153.

The second 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 and (2) add a bypass to the high flow trip of the "B" Isolation Condenser when initiating the alternate shutdown panel. The licensee is proposing to increase the value of the high drywell pressure trip setting in Table 3.1.1 of the TS. This applies to reactor scram, core spray initiation, containment spray initiation, containment isolation, automatic reactor vessel depressurization, Reactor Building isolation and the Bases in Section 3.1 of the TS for the table. 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. This second amendment is in accordance with the licensee's application for amendment dated September 9, 1986, for TSCR 147.

Before issuance of either proposed license amendment, the Commission will have made findings required by the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations.

The Commission has made a proposed determination that both amendment requests involve no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendments would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

The basis for this proposed determination for the first proposed amendment is the following. The first amendment proposes to revise the period of time when a protective instrumentation channel listed in Table 3.1.1 may be made inoperable without tripping its associated trip system. In the existing TS, these channels may be taken out of service to perform tests and calibrations for up to 1 hour per month without tripping the associated trip system. The proposed amendment would change that to 2 hours only for each required TS surveillance. The frequency of TS-required surveillances is listed in Table 4.1.1 of the TS. The proposed amendment should be more restrictive than the existing TS for 22 out of the 27 separate parameter (e.g., drywell pressure, reactor water level low function, APRM level) channels listed in Table 4.1.1. Each parameter channel is actually four separate independent channels measuring the same parameter.

The licensee expects that the time needed for the analog trip system channels in the reactor protection system to be taken out of service for TS required tests and calibration and then returned to service is greater than an hour. Therefore, with the existing TS, every time the analog channel is taken out of service for TS-required tests and calibration, the channel may have to have its associated trip system be placed in the tripped condition while the channel is still under surveillance. These analog channels result from a modification to the reactor water level instrumentation system in the present outage which replaced digital sensing devices with an analog trip system and this situation did not exist before.

The channel performs its function by causing its associated trip system to trip in response to a safety setpoint being exceeded as, for example, high drywell pressure. A channel is tripped because it is inoperable, that is, not capable of actuating its associated trip system or because it is out-of-service too long and again is not capable of actuating its associated trip system. A channel is not tripped when it is taken out of service for tests or calibrations because putting it in the tripped condition increases the change of spurious or inadvertent trips or scrams and thus unnecessary challenges to safety systems. In addition, the channel was operable prior to being taken out of service and there is no reason to believe the other channels are inoperable. Placing a channel in the tripped condition when it is inoperable also increases the chance of spurious or inadvertent trips or scrams but the fact the channel was inoperable may mean the other channels are more subject to the chance for being inoperable.

In addition, placing a reactor water level low low function channel in the tripped condition causes all four core spray pumps to unnecessarily start up. This channel is one of the analog channels discussed above. The proposed amendment would prevent starting up these pumps during required TS surveillance on these channels.

Monthly surveillance testing is necessary to provide a high degree of reliability for the automatic actuation circuits of the Reactor Protection and Engineered Safety Feature Systems. In order to test the actuation circuit completely, it must be made inoperable but not tripped. Tripping the channel rather than making it inoperable during the required surveillance testing would increase the likelihood of spurious scrams or unnecessary challenges to safety systems. Also, given the tested reliability of the operational instrument, an increase in out of service time from 1 hour to 2 hours will have a negligible effect on channel failure rate.

Additionally, the Standard Technical Specifications for General Electric Boiling Water Reactors, NUREG-0213, specifies a 2-hour limit for TS required surveillance.

For this first proposed amendment submitted in the licensee's letter dated September 5, 1986, the proposed changes should not involve a significant hazards consideration because operation of Oyster Creek in accordance with these changes would:

(1) not involve a significant increase in the probability or consequences of an accident previously evaluated. The protective instrumentation channels and the condition of the associated channel trip system during channel TS surveillance does not change the probability of an accident. The channels and the channel system respond to off normal conditions (i.e., conditions exceeding safety setpoints) to prevent or respond to accidents or accident conditions. The channels and channel trip systems are not initiators of accidents but systems to act to prevent or respond to accidents. The channels and channel trip do not change the consequences of an accident because only one of the four independent channels measuring the same parameter is taken out of service at a time for TS surveillance. Therefore, three channels remain for each parameter to perform the functions of responding to changes in that parameter.

(2) not create the possibility of a new or different kind of accident from any previously analyzed. The amendment does not create the possibility of a new or different kind of accident because only one of the four channels measuring a parameter is allowed to be out-of-service without the channel trip system in the tripped condition. By the design of the trip logic circuitry, two of the other three parameter channels will respond to changes in the parameter being measured and actuate their trip system.

(3) not involve a significant reduction in a margin of safety. This amendment either (1) decreases the period of time that a channel may be made inoperable with its associated trip system not placed in the tripped condition or (2) increases it insignificantly. For those channels where the period of time is increased, there are other channels following independent parameters for which the period of time should be decreasing with the amendment. The typical increase for a channel is from 1 hour per month to 2 hours per month which is an increase of 12 hours per year or 0.137% per year. The worst increase is for one channel and is an increase of 1 hour per 3 days or 122 hours per year. This is an increase of 1.39% per year. These are worst case estimates since the actual time the channel may be out-of-service for TS tests and calibrations should be less than the 2 hours.

The basis for the proposed determination for the second proposed amendment is the following. The second amendment proposes to (A) increase the setpoint for drywell pressure channel to actuate its associated trip system and cause reactor scram, core spray, containment spray, containment isolation, reactor vessel depressurization and Reactor Building isolation and (B) add a bypass to the high steam line flow and high condensate return line flow for "B" Isolation Condenser isolation for when the alternate shutdown panel is initiated. These two lines are lines to and from the Isolation Condenser. 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 for a postulated Appendix R event. This event involves a fire. For this event, the drywell cooling fans are assumed lost due to the fire, and the reactor is cooled by the Isolation Condenser with no feedwater flow. Thus, it is essential that the Automatic Depressurization System (ADS) logic does not actuate to further reduce reactor water level. The analysis for this event concluded that the drywell pressure could reach 1.9 psig which would exceed the current high drywell pressure instrument setpoint of 1.85 psig. 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 current instrument setpoint for high drywell pressure and would result in an inadvertent initiation of ADS. In order to prevent 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 was requested for the drywell pressure.

For evaluating the acceptability of increasing the drywell pressure TS limit to 3.5 psig, the effect of the increased TS limit on anticipated plant operational occurrences and accidents was evaluated. 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 subsequently 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 factors.

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 pressure difference and associated time delay is less.

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 period of 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 earlier regarding the negligible delay in high drywell pressure indication associated with a 1.1 psig TS limit increase are applicable. 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. Since 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 normal mode of operation.

The following is related to the proposed bypass for "B" Isolation Condenser isolation. An 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 loss of control room function due to fire damage in the cable spread rooms. 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 isolation condenser, the design includes a bypass of the trip function upon initiation of the alternate shutdown panel.

The high flow trip function is provided to isolate the system in the event of a line break outside primary containment. The occurrence of a fire requiring initiation of the alternate shutdown panel in conjunction with a line break accident is not considered a credible event. The alternate shutdown panel is initiated through transfer switches which are key locked and alarmed in the control room to prevent inadvertent actuation. Single failure of the switch will not preclude operation of the isolation condenser high flow trip in the event of a line break accident.

The design of the alternate shutdown system including bypassing the high flow trip function was reviewed and approved by the Nuclear Regulatory Commission in its Safety Evaluation dated March 24, 1986.

Based upon the above discussion for the second proposed amendment, the proposed change should not involve significant hazards consideration. In summary, it has been determined that the proposed amendment would:

(1) not involve a significant increase in the probability or consequences of an accident previously evaluated;

(a) The proposed change to the high drywell pressure TS limit does not alter the probability of any previously evaluated accident because the TS limit is not an initiator of an accident. For each case analyzed, the delay in high drywell pressure indication because of the higher setpoint had minimal or no effect on the accident severity.

(b) The proposed bypass of the high flow trip is not an initiator of an accident and it is used only in response to a severe fire; therefore, it does not change the probability of an accident previously evaluated. The proposed bypass is used to assure that an isolation condenser would be available when needed during a severe fire and has been reviewed and approved by the NRC staff and, therefore, it should not increase the consequences of an accident previously evaluated.

(2) not create the probability of a new of different kind of accident from any accident previously evaluated;

(a) The proposed increase to the high drywell pressure setpoint only involves a small increase to a trip setting. This results in minimal or no effect on when automatic protective actions are assumed to be initiated in accident analyses. It also does not involve a change of any of the limiting safety system settings listed in Section 2.3 of the Oyster Creek TS.

(b) Bypassing the isolation condenser high flow trip occurs only during initiation of the alternate shutdown panel. This bypass is to assure the operation of an isolation condenser when it may be needed.

(3) not involve a significant reduction in a margin of safety;

(a) The proposed increase in the high drywell pressure setpoint has minimal or no effect on the severity of the accidents analyzed.

(b) The proposed bypass of the high flow trip is to assure operation of an isolation condenser when it may be needed and single failure of the switch to initiate the alternate shutdown panel will not preclude operation of the isolation condenser high flow trip in the event of an isolation condenser line break accident. The occurrence of a severe fire requiring initiation of the alternate shutdown panel and then an isolation condenser line break before the alternate shutdown panel is in operation is not considered sufficiently credible to design for. This was reviewed and approved by the NRC staff.

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination. The Commission will not normally make a final determination unless it receives a request for a hearing.

Written comments should be addressed to the Rules and Procedures Branch, Division of Rules and Records, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, and should cite the publication date and page number of this FEDERAL REGISTER notice. Copies of comments received may be examined at the NRC Public Document Room, 1717 H Street, NW, Washington, D.C.

By October 17, 1986 , the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written petition for leave to intervene. Request for a hearing and petitions for leave to intervene shall be filed in accordance with the Commission's "Rules of Practice for Domestic Licensing Proceedings" in 10 CFR Part 2. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of hearing or an appropriate order.

As required by 10 CFR §2.714, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following factors: (1) the nature of the petitioner's right under the Act to be made a party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to fifteen (15) days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

Not later than fifteen (15) days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter, and the bases for each contention set forth with reasonable specificity. Contentions shall be limited to matters within the scope of the amendment under consideration. A petitioner who fails to file such a supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses.

If a hearing is requested, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held.

If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment.

If the final determination is that the amendment involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

Normally, the Commission will not issue the amendment until the expiration of the 30-day notice period. However, should circumstances change during the notice period such that failure to act in a timely way would result in derating or shutdown of the facility, the Commission may issue the license amendment before the expiration of the 30-day notice period, provided that its final determination is that the amendment involves no significant hazards consideration. The final determination will consider all public and State comments received. Should the Commission take this action, it will publish a notice of issuance and provide for opportunity for a hearing after issuance. The Commission expects that the need to take this action will occur very infrequently.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attn: Docketing and Service Branch, or may be delivered to the Commission's Public Document Room, 1717 H Street, NW Washington, D.C., by the above date. Where petitions are filed during the last ten (10) days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at (800) 325-6000 (in Missouri (800) 342-6700). The Western Union operator should be given Datagram Identification Number 3737 and the following message addressed to John A. Zwolinski, Director, BWR Project Directorate #1, Division of BWR Licensing: petitioner's name and telephone number; date petition was mailed; plant name; and publication date and page number of this FEDERAL REGISTER notice. A copy of the petition should also be sent to the Office of the General Counsel-Bethesda, U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, and to Ernest L. Blake, Jr., Shaw, Pittman, Potts and Trowbridge, 1800 M Street, NW, Washington, D.C. 20036, attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for hearing will not be entertained absent a determination by the Commission, the presiding officer or the presiding Atomic

Safety and Licensing Board, that the petition and/or request should be granted based upon a balancing of factors specified in 10 CFR 2.714(a)(1)(i)-(v) and 2.714(d).

For further details with respect to this action, see the applications for amendment which are available for public inspection at the Commission's Public Document Room, 1717 H Street, NW, Washington, D.C., and at the Local Public Document Room located at the Ocean County Library, 101 Washington Street, Toms River, New Jersey 08753.

Dated at Bethesda, Maryland, this 12th day of September 1986.

FOR THE NUCLEAR REGULATORY COMMISSION

John A. Zwolinski, Director
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Division of BWR Licensing