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Perry Nuclear Power Plant
Docket No. 50-440
Supplement to a License Amendment Request: Revision of Main
Steam Line Leakage Requirements and Elimination of the Main Steam
Isolation Valve Leakage Control System (TAC No. M96931)

Ladies and Gentlemen:

A supplement to a license amendment request for the Perry Nuclear Power Plant (PNPP) is provided. The amendment involves revision of the Main Steam Line leakage requirements and elimination of the Main Steam Isolation Valve Leakage Control System. This amendment was submitted to the Nuclear Regulatory Commission (NRC) by letters dated August 27, 1996 (PY-CEI/NRR-2076L), and July 22, 1998 (PY-CEI/NRR-2299L). Revised Accident Source Term (RAST) dose calculations provide a primary justification for this amendment.

A PNPP-specific meeting was held with the NRC staff on September 15, 1998, and an industry meeting between the NRC and the Nuclear Energy Institute (NEI) was held on October 1, 1998. At these meetings and in subsequent telephone discussions, resolution of several staff issues has been achieved. Summaries of these items are provided in Attachment 1. Attachment 2 provides an update to two of the commitments provided in the letters dated August 27, 1996, and July 22, 1998, regarding the use of the containment spray system and a backup method for pH control of the suppression pool.

Attachment 3 provides the two calculations that have been revised as a result of issue resolutions with the NRC staff. These calculations are considered to be Proprietary. Non-Proprietary versions of these calculations can be provided at NRC request. Proprietary affidavits from Polestar Applied Technology, Inc., were submitted for these calculations as part of the letters dated August 27, 1996, and July 22, 1998. Pursuant to 10 CFR 2.790(b)(1), it is requested that Attachment 3 be withheld from public disclosure.

Other than the updated commitments described in Attachment 2, this letter does not contain regulatory commitments. If you have questions or require additional information, please contact Mr. Henry L. Hegrat, Manager - Regulatory Affairs, at (440) 280-5606.

Very truly yours,

APO1 1/1

Rn Schrander

For Lew W. Myers

220007

Attachments

cc: NRC Project Manager
NRC Resident Inspector
NRC Region III
State of Ohio

Attachment 3 contains 2.790(a)(4) PROPRIETARY information. Upon removal of Attachment 3, the remainder of this letter and its Attachments may be disclosed.

ltr encl
1 1w/o prop

I, Robert W. Schrauder, being duly sworn state that (1) I am Director, Perry Nuclear Engineering Department, of the FirstEnergy Nuclear Operating Company, (2) I am duly authorized to execute and file this certification as the duly authorized agent for The Cleveland Electric Illuminating Company, Toledo Edison Company, Duquesne Light Company, Ohio Edison Company, and Pennsylvania Power Company, and (3) the statements set forth herein are true and correct to the best of my knowledge, information and belief.

R W Schrauder

Robert W. Schrauder

Sworn to and subscribed before me, the 18th day of January, 1999.

Brenda Alvord

My commission expires 8-5-2001

Changes To The Revised Accident Source Term (RAST) Calculations

The calculations included in Attachment 3 were revised to reflect resolution of several items with the NRC staff. Note that these issues are specific to the Perry Nuclear Power Plant (PNPP). Several of these items are expected to receive additional attention as part of the generic regulatory guide development in support of the current rulemaking effort. It is possible that if PNPP revises RAST calculations in support of future applications, the PNPP values may change in such future calculations as a result of the generic industry discussions. The primary changes are summarized below. Further details on these changes are described in the attached calculations.

- The sweepout flow rate from drywell to containment during the fission product release phase of the loss of coolant accident (LOCA) event (approximately the first two hours) is assumed to be 3000 cubic feet per minute (cfm) in the thermal-hydraulic calculation. This was reduced by more than half from the previous calculation, from ≈6200 cfm. Although there are a number of mechanisms that would produce sweepout flow rates greater than or equal to the 6200 cfm sweepout rate, this change was made in order to increase the calculated dose from the Main Steam Lines during the first two hours of the event.
- After two hours, the drywell and the unsprayed region of the containment are assumed to be well-mixed. The ECCS injection function is assumed to be recovered at the two hour point, which is an integral assumption for application of the revised accident source term. The previous calculations assumed that when the Emergency Core Cooling Systems (ECCS) began injecting, the resultant steam generation spike would sweep essentially all of the source term out of the drywell and into the unsprayed region of the wetwell (the containment). This change to a well mixed volume was made so that the mitigation techniques for the leakage from both the containment and the drywell would be more equally challenged in the time frame after the ECCS injection recovery at two hours.
- The Main Steam Line aerosol removal efficiency (the ability of the steam lines to retain aerosol fission products) was assumed to be slightly reduced as compared to the assumptions in the previous calculations. The aerosol removal efficiency in the new calculation is equivalent to an increase in aerosol penetration of 10 percent. This was done to further increase the calculated dose from the Main Steam Line pathway.
- The dispersion values (also known as Chi/Q or X/Q) used in the Control Room dose calculations were revised to utilize more conservative values. The current PNPP licensing basis Chi/Q's are based on a PNPP-specific study that was included as Attachment 5 of the August 27, 1996 letter. Following discussions with the NRC, it was decided that for these revised accident source term calculations, the values used would be somewhat more conservative, especially for the first several hours following the event. Note that in the interim period until any license amendments based on the revised accident source term calculations are made effective, the current calculations based on the existing Chi/Q values are considered to remain valid.

Integrated Assessment

During the meeting with the NRC staff on September 15, 1998, it was requested that an examination be performed on a variety of subjects related to the potential impact of applying the revised accident source term at the Perry Nuclear Power Plant. This was described using the term "integrated assessment".

To determine the implications of the Perry revised accident source term on aspects of plant design other than the radiological design basis accident (DBA) LOCA, an integrated assessment has been performed. This assessment addressed environmental qualification (EQ) of equipment, vital area access, post accident sampling system (PASS) access, and probabilistic risk assessment (PRA) impact. Scoping evaluations in eight areas were performed:

1. Annulus exhaust gas treatment system (AEGTS) filter train EQ dose
2. Control room emergency recirculation filter train EQ dose
3. EQ dose adjacent to main steam line between inboard and outboard main steam isolation valves (MSIVs)
4. Emergency core cooling system (ECCS) equipment EQ dose
5. Drywell atmosphere EQ dose
6. Vital area access dose
7. PASS dose
8. PRA impact of technical specifications changes associated with the revised accident source term

The first five areas were selected to provide a representative sample of PNPP EQ equipment, and to address equipment where there could be differences in the doses from the revised accident source term vs. the TID-14844 source term (the revised accident source term demonstrated that there would be a different percentage of the various chemical forms of radionuclides). Vital area and PASS access doses were evaluated since they involve specific post-LOCA design basis operator actions outside the control room. Finally, PRA impact was addressed to provide a risk-informed perspective for this revised accident source term license amendment application at PNPP.

Specific results are as follows:

- The long term (180 day) AEGTS filter train plenum EQ dose from the revised accident source term was evaluated to be below that from TID, and well below the qualification envelope for radiation sensitive components in the vicinity of the plenum.

- The long term (180 day) control room emergency recirculation filter train EQ dose was evaluated to be well below the qualification envelope of radiation sensitive components in the vicinity of the filter train.

- The long term (180 day) dose adjacent to the main steam line (from aerosol deposition in the steam line) was evaluated to be well below the qualification envelope of radiation sensitive components in the vicinity of the steam line.

- The long term (180 day) ECCS equipment EQ dose based on revised accident source term was evaluated to be below that from TID, and well below the qualification envelope for radiation sensitive ECCS components, due to the fact that the Perry TID-based ECCS equipment EQ used a 50% cesium source term.

- The long term (180 day) drywell atmosphere EQ integrated dose was evaluated to be a factor of 3 less than the qualification envelope of the minimally qualified radiation-sensitive component in the drywell.

- The revised accident source term dose to personnel for vital area access was evaluated to be below that based on TID.
- The revised accident source term dose to personnel for PASS access was evaluated to be below that based on TID.
- Based on the Perry Individual Plant Examination (IPE), there is essentially zero impact on Perry core damage frequency and large early release frequency from the technical specification changes associated with the application of the revised accident source term (revision of main steam line leakage requirements and elimination of the Main Steam Isolation Valve Leakage Control System).

Based on these results, the following conclusions may be drawn on the integrated assessment:

1. The existing TID-based environmental qualification envelopes provide adequate margin for the revised accident source term for EQ. Thus, the application of the revised accident source term does not necessitate changes to the existing EQ design basis for plant components.
2. The existing TID-based radiological design bases for vital area access and PASS access provide adequate margin, and changes to their design basis are not necessary.
3. The risk impact of Perry plant changes associated with application of the revised accident source term is negligible.

Consideration Of Events Other Than A Large Break LOCA

During the meeting with the NRC staff on September 15, 1998, it was also requested that consideration be given to the potential impact of applying the revised accident source term to several events other than the large break LOCA at the Perry Nuclear Power Plant. This was described as a scoping level evaluation, and quantitative/detailed calculations were not necessary to meet the intent of these evaluations.

In determining the impact of the PNPP revised accident source term on such other events, consideration was given to several design basis accidents and transients. Also, a beyond-design-basis event was considered, although such severe accident considerations are considered to be outside the envelope of items formally under NRC review for this license amendment.

Design Basis Events

The events considered to be design-basis accidents at PNPP are summarized in Updated Safety Analysis Report Table 15.0-3, which also identifies the amount of failed fuel that is assumed for each of the listed accidents. Most of these accidents do not result in failed fuel, and therefore are not discussed below, since application of the revised accident source term is clearly not applicable to those design-basis accidents. As noted in the Table, even the design-basis large and small break LOCAs discussed above are not postulated to result in failed fuel, since the normal plant operating limits such as the Average Planar Linear Heat Generation Rate (APLHGR) limits and the design of the plant are such that post-LOCA fuel temperatures will not exceed 2200°F. However, regulatory guidance directs that a source term from a severe accident (previously the TID-14844 source term, currently the NUREG-1465 source term) be utilized as the design-basis LOCA source term, even though such a severe accident source term would not actually be released from a true design-basis event.

Fuel Handling Accident

The fuel handling accident (FHA) both inside and outside of the containment was considered, and the conclusion was that application of the revised accident source term to the PNPP FHA's will result in doses below the existing TID-based doses for the FHA's. Thus, changes to the existing analyses are not considered necessary at this time.

Small Break Loss of Coolant Accident

A small break LOCA was also examined in a qualitative comparison with the newly performed design-basis revised accident source term (RAST) calculations for the large break LOCA. The radiological DBA for Perry is a large LOCA with the NUREG-1465 source term release fractions entering the drywell over a 2 hour period after the initiating event. A small LOCA also has a pathway from the reactor coolant system (RCS) directly to the drywell, but would likely have slower core degradation than a large LOCA.

For small LOCA, core degradation would occur over a period of several hours, the exact time depending upon the size of the break. Any delay in core damage vs. the 2 hour release duration for large LOCA will reduce dose due to radiological decay of the fission products. Also, delay in core damage will increase fission product residence time in the RCS, thus increasing aerosol retention in the RCS and decreasing release to the drywell atmosphere compared to large LOCA. Steaming rate during fission product release for a small LOCA would likely not be any smaller than assumed in the Attachment 3 calculations for a large LOCA, since for small LOCA there would be more residual water in the reactor vessel at the beginning of core damage. Finally, small LOCA fission product release duration may be extended vs. large LOCA, which would decrease the maximum 2 hour dose for the same release fraction. For these reasons, integrated airborne fission product concentration for small LOCA was evaluated to be bounded by that from large LOCA.

Control Rod Drop Accident

The changes being proposed by this license amendment do not impact this accident. The Control Rod Drop event has been previously reanalyzed to not require closure of the Main Steam Isolation Valves based on a high radiation signal. Therefore, the analyses do not credit any specific leakage rate for the MSIVs. The Main Steam Isolation Valve Leakage Control System has never been credited for any event other than the design-basis LOCA. Therefore, the analyses for the Control Rod Drop Accident were not revised as a result of applying the revised accident source term.

Loss of Offsite Power, or Various Transients

Other events, such as a loss of offsite power (LOOP) or various transients, are already analyzed using design basis guidelines and shown to not result in core damage. Regulatory guidance does not direct that the severe accident source term be postulated for events such as the LOOP or other transients. Therefore, their current analyses are not changed as a result of applying the revised accident source term to PNPP.

Other Events Required To Be Analyzed By Regulation

Regulatory guidance also does not direct that the severe accident source term be postulated for events that have been required to be analyzed per regulations imposed in addition to the normal design-basis events. These "licensing basis" events are the Anticipated Transients Without Scram (ATWS) and the Station Blackout (SBO) events. These were imposed by 10 CFR 50.62 and 50.63, respectively. These events have been previously analyzed for PNPP and shown to not result in significant core damage or offsite releases. The application of the revised accident source term to PNPP does not impact the analyses of the ATWS or SBO events in any way.

Beyond-design-basis events

In the meeting with the NRC staff on September 15, 1998, it was requested that an event be considered in which the release pathway for a revised accident source term from the reactor pressure vessel is through the Safety Relief Valves rather than through a piping break leading to the drywell. This would be an "isolated vessel" event that had multiple failures such that it led to a core damage event. This would be an event such as a loss of AC power event that progressed much further than the Station Blackout event required by the regulation discussed above. This is discussed here to provide completeness, but is considered to be a severe accident rather than a DBA or a licensing basis event, and therefore is outside the envelope of items formally under NRC review for this license amendment.

For an isolated vessel event, the onset of core damage would be delayed in comparison to the large break LOCA event, for up to several hours. This would provide for additional decay of the radionuclides, and decrease the maximum 2 hour dose due to the slower release.

In such an event, the main fission product release path would be through the safety relief valves (SRVs) into the suppression pool. Thus the drywell would be bypassed (for a LOCA, drywell lambda from sedimentation and sweepout is about 1 per hour which is a decontamination factor (DF) of about 7 over a 2 hour period). However, pool scrubbing occurs for this SRV pathway. The pool scrubbing DF will vary, but is unlikely to be less than a factor of 10. Thus the source term is subjected to a higher DF for this pathway than via the drywell. Further, any drywell atmosphere source term would be from flow back from containment, and would receive further dilution from the drywell atmosphere, so the drywell source for leakage is reduced compared to that for a LOCA.

There is also the RCS pathway via the intact vessel and steam lines to the MSIVs. For this pathway, however, it is considered that a number of factors would mitigate the release such that the integrated doses for the beyond-DBA isolated vessel event would be bounded by the doses calculated for the DBA large break LOCA. These factors include: the increased time for decay noted above; the reduction in fission product concentration available for release through the MSIVs due to the fission product gas and vapor flow through the SRVs to the suppression pool; the stratification of cooler gas in the 26 foot vertical section of steam line between the RPV and the inboard MSIV which will further delay the transport of hot fission product gas and aerosol to the inboard valve; the longer residence time in the RCS which will increase aerosol retention; and retention of aerosol in the safety-related piping section between the outboard MSIV and the third isolation valve (Main Steam Shutoff Valves) as well as in the non-safety main steam piping downstream of the Main Steam Shutoff Valve (neither of which are credited in the DBA analysis).

Commitment Update

Containment Spray

In the letters dated August 27, 1996, and July 22, 1998, the following commitment was included:

- **Based on a high radiation signal in the Control Room, the Containment Spray system would be operated post-LOCA for up to 24 hours (previous analyses assumed 6 hours of spray operation), in order to scrub released radionuclides from the containment atmosphere and into the suppression pool, and thus reduce the post-LOCA off-site and Control Room dose.**

In the interim period since this commitment was made in 1996, work has progressed on additional plant operator guidance for events that might result in large releases of radioactivity from the nuclear fuel, such as are postulated in the Revised Accident Source Terms of Reference 1. This work has been done under the Severe Accident Management (SAM) Mitigation effort. Use of containment sprays for post-accident dose mitigation is a part of this effort. The new guidelines direct that containment sprays be initiated based on readings from the NUREG-0737 containment high range radiation monitor rather than the Control Room radiation monitor. This is considered to be a more appropriate radiation monitor to use. Therefore, the commitment quoted above is being revised to reflect the change in the identified radiation monitor from the Control Room monitor to the Containment monitor. [Note: The Alarm Response Instruction (ARI) for the Control Room radiation monitor will, however, continue to direct use of containment sprays should that monitor indicate high Control Room dose rates.]

The commitment also includes the phrase "for up to 24 hours". This phrase is intended to address two concepts.

First, the dose calculations assume the sprays are run for the first 24 hours, then are suspended. This is the most important time period for scrubbing of radiation down into the suppression pool. However, in an actual event, spray use would not necessarily be suspended at 24 hours, if appropriate conditions for their use still existed. Therefore, the phrase "up to" is not intended to be interpreted as a commitment to stop using sprays after 24 hours.

Second, the phrase "up to" is intended to mean that in an actual event, the sprays will be run when it is appropriate, and not necessarily the entire time during the first 24 hours of a LOCA. This does not invalidate the assumptions in the dose calculations, however. The accident guidance to operators is written to be symptom based, rather than event based. This was a lesson learned from the Three Mile Island accident. Most postulated LOCAs will not result in large radiation releases (in fact the PNPP design basis LOCA results in peak clad temperatures less than 2200°F). Therefore, it would not be appropriate to run containment sprays for 24 hours following such an event. As noted above, it is more appropriate to base spray use on the containment radiation monitor reading, which would indicate the need for spray if the "two hours with no Emergency Core Cooling System (ECCS) cooling" event (as postulated in the RAST calculations) were to actually occur. Another critical factor in spray use is containment pressure. Use of the sprays will work to reduce containment pressures, due to steam condensation and the containment heat removal function that they provide. If a high radiation signal is present from the containment radiation monitor and pressures are elevated in containment, the instructions direct that the sprays be run. However, if containment pressure reduces to near zero and use of the sprays is terminated by the operators, this does not have an adverse impact on offsite doses (or the dose calculations) since the driving pressure for containment and MSIV leakage has been eliminated. The dose calculations assume that the maximum allowable leakage (La) corresponding to the peak post-accident pressure (P_a) remains during the entire 24 hour period, so if containment pressure actually reduces to substantially less than P_a, a reduction in leakage and the resultant offsite doses will follow. If containment pressure increases again, and the high radiation signal is still present, the instructions would again direct use of the sprays.

These SAM guidelines became effective in December 1998.

Based on the above discussion, the commitment on use of containment sprays is being revised by PNPP to read:

- **Based on a high radiation signal in the Containment, the Containment Spray system would be operated post-LOCA for up to 24 hours (previous analyses assumed 6 hours of spray operation), in order to scrub released radionuclides from the containment atmosphere into the suppression pool, and thus reduce the post-LOCA off-site and Control Room dose.**

Backup Method of pH Control

In the letters dated August 27, 1996, and July 22, 1998, the following commitment was included:

- **A backup method for pH control will be developed for use if post-accident suppression pool sampling identifies that the primary pH control method is not being effective.**

The following provides further information on the primary and backup methods of pH control.

As noted in the previous letters, the pH level of the suppression pool will be raised to 7 or above post-LOCA, and then maintained at 7 or above. The method for pH control will use the existing Standby Liquid Control (SLC) system. This is a redundant, safety-related, Technical Specification controlled system, which can inject a boron solution into the reactor vessel. This buffers the acids that are postulated to eventually migrate into the suppression pool due to radiolytic decomposition of compounds such as the insulation on cables in containment due to high postulated radiation levels. In the event of a large break LOCA, during the vessel reflood period, the boron solution would spill from the break, fill the drywell, and return over the drywell weir wall back into the suppression pool. This sets up a recirculation path back to the ECCS pumps, and provides for equal distribution of the buffer solution around the circumference of the suppression pool.

The commitment listed above provides for development of a backup method of providing buffer solution to the suppression pool if the primary pH control method (injection of SLCS into the reactor vessel) was not proving effective. This commitment was made to address the case that might occur if a LOCA has occurred very high up on the reactor vessel. In such an event, operators could recover water level over the core without necessarily spilling it out of the break, and therefore the buffer solution would not get to the suppression pool (if the break is more than 215 inches above the top of the active fuel). In this case, the vessel water is highly buffered, but the suppression pool could eventually drop below a pH of 7.

Therefore, a backup method of getting the buffer solution to the suppression pool will be through direction to operators to establish a flow path directly to the suppression pool from the reactor vessel, through the Safety Relief Valves (SRVs). This is similar to the already proceduralized, safety related, "Alternate Shutdown Cooling" method, which involves opening Safety Relief Valves (SRVs) and flooding up the reactor to spill the buffered water down to the suppression pool through the SRV spargers. This can be accomplished from the Control Room. The ability of the SRVs to perform in such a mode was the subject of a testing program which was reviewed and approved in Supplement 7 to the NRC Safety Evaluation Report for PNPP (Confirmatory Issue 7, resolved in Section 3.9.3.2.1). There are SRV spargers distributed around the entire suppression pool. There are multiple emergency instruction steps which would have already directed depressurization of the reactor pressure vessel in an event with low water levels and fuel damage on the order of magnitude being considered here. Depressurization instructions require opening of at least eight of the 19 SRVs if available (preferably the Automatic Depressurization System (ADS) SRVs). The SRVs can be opened and maintained open from control switches in the Control Room, and the ADS valves are powered from DC supplies. Additional guidance is proposed to be added to the SAM guidelines to raise water level to permit it to flow out the open SRVs to enhance suppression pool buffering if sampling of the pool shows that the normal ph control method is not proving effective. The use of multiple SRVs for this flow would provide for good mixing of the buffered water with the suppression pool.