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Fred Dacimo
Site Vice President
Administration

December 22, 2004

Re: Indian Point Unit 3
Docket No. 50-286
NL-04-162

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

SUBJECT: Reply to RAI regarding Alternate Source Term License Amendment and Transmittal of Supplemental Information (TAC MC3351)

References:
1. NRC letter dated December 6, 2004; "Request for Additional Information Regarding Amendment Application for Alternate Source Term". [TAC NO. M3351]
2. Entergy letter to NRC (NL-04-068) dated June 2, 2004; "Proposed Change to Technical Specifications Regarding Full Scope Adoption of Alternate Source Term".

Dear Sir;

Entergy Nuclear Operations, Inc. (ENO) is providing a response to the NRC request for additional information (RAI) in Reference 1 regarding the proposed license amendment request for adoption of Alternate Source Term (Reference 2) for Indian Point 3 (IP3). The responses to questions are provided in Attachment 1. Attachment 2 contains an affidavit from Polestar Applied Technology, Inc. (Polestar) requesting the withholding of proprietary information (a calculation referenced in Attachment 1) pursuant to the provisions of 10 CFR 9.17(a)(4). The proprietary version of this calculation is contained in Attachment 3 and the non-proprietary version is in Attachment 4.

Commitments made by this submittal are identified in Attachment 5. If you have any questions or require additional information, please contact Mr. Patric W. Conroy at 914-734-6668.

AP01

I declare under penalty of perjury that the foregoing is true and correct. Executed on December 22, 2004.

Sincerely,

E. O'Donnell

for Fred R. Dacimo
Site Vice President
Indian Point Energy Center

cc: Mr. Patrick D. Milano, Senior Project Manager
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ATTACHMENT 1 TO NL-04-162

REPLY TO NRC REQUEST FOR ADDITIONAL INFORMATION REGARDING PROPOSED LICENSE AMENDMENT REQUEST FOR ADOPTION OF ALTERNATE SOURCE TERM AT INDIAN POINT 3

ENTERGY NUCLEAR OPERATIONS, INC
INDIAN POINT NUCLEAR GENERATING UNIT 3
DOCKET 50-286

Request for additional information
NRC letter dated December 9, 2004 (TAC NO. M3351)

Attachment 1 – Proposed Changes

1. The proposed re-definition for dose equivalent iodine isotope 131 (I^{131}) allows the use of the committed effective dose equivalent (CEDE) dose conversion factors for either the submersion, inhalation and ingestion pathways. The definition must be modified to indicate that it is only the inhalation pathway CEDE dose conversion factors.

   Response: The proposed definition refers to Table 2.1 of EPA Federal Guidance Report No. 11 which is entitled "Exposure-to-Dose Conversion Factors for Inhalation". Thus no change is required.

2. What is the basis for including ^{130}I in the calculation of dose equivalent ^{131}I?

   Response: Historically, the only iodine isotopes that have been considered in the accident dose analyses are 131, 132, 133, 134, and 135. The definition of dose-equivalent I-131 was modified to include I-130 because I-130 is included in the accident dose analyses. The inclusion of I-130 in the accident dose analyses is due to the determination that it has a greater dose significance in the accident analyses than I-134 does.

3. It has been proposed that the testing requirements for the Fuel Storage Building Emergency Ventilation System be deleted. It appears that the basis for its removal is the fact that it has been determined that a fuel-handling accident occurring within containment results in acceptable offsite and control room operator doses without the assumption of containment integrity and without credit for filtration. This filtration system has been utilized to reduce the release of effluents during refueling operations. Section II.D of Appendix I to 10 CFR Part 50 requires licensees to include in their radwaste systems all items of reasonably demonstrated technology that, when added to the system sequentially and in order of diminishing cost-benefit return can for a favorable cost-benefit ratio effect reductions in dose to the population reasonably expected to be within 50 miles of the reactor. Your Appendix I analysis assumed filtration of the effluents during fuel-handling operations. Provide your analysis which demonstrates that removal of the Fuel Storage Building Emergency Ventilation System is in compliance with Section II.D of Appendix I.

   Response: The request for additional information indicates that removal of testing requirements for the FSB emergency ventilation system requires a cost benefit analysis utilizing the criteria of 10 CFR 50 Appendix I to justify removal. Entergy has requested removal of the requirements for testing the FSB emergency ventilation system from the TS and will relocate them to Chapter 9 of the FSAR. Once relocated, the 10 CFR 50.59 process can revise the testing requirements for consistency with 10 CFR 50 Appendix I criteria. A precedent exists for this. The testing requirements for the FSB emergency ventilation system were removed from the Indian Point Unit 2 TS when adopting the alternate source term and relocated to the FSAR. They were subsequently revised to meet 10 CFR 50 Appendix I requirements utilizing the guidance of Regulatory Guide 1.140 insofar as they applied. These requirements insure structural integrity and performance capabilities are maintained consistent with GDC 61. Because charcoal in the FSB emergency ventilation system is not credited for the fuel handling accident, the requirements of 10 CFR 50.36(c)(2)(ii) no longer require a TS. These requirements state:

> "ii) A technical specification limiting condition for operation of a nuclear reactor must be established for each item meeting one or more of the following criteria:
> (A) *Criterion 1.* Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
> (B) *Criterion 2.* A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
> (C) *Criterion 3.* A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
> (D) *Criterion 4.* A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety."

4. If containment integrity is not established for a fuel-handling accident and the Fuel Storage Building Emergency Ventilation System is not operating, explain how the requirements of General Design Criteria (GDC) 60, 61 and 64 are met during these fuel handling operations.

Response: It is noted that the NRC staff has already approved refueling operations at Indian Point 3 with the containment open and without credit for filtration of releases from a postulated Fuel-Handling Accident (FHA) inside containment or in the Fuel Storage Building (NRC Safety Evaluation, "Issuance of Indian Point 3 Amendment 215 for Selective Adoption of Alternate Source Term," dated March 17, 2003).

The Fuel Storage Building Emergency Ventilation System would be expected to be in operation during fuel-handling operations but it is not required to be in operation to mitigate the consequences of the FHA.

The RAI requests clarification regarding how the requirements of 10 CFR 50, Appendix A, General Design Criteria 60, 61, and 64 are met during fuel handling operations if containment integrity is not established and the FSB emergency ventilation system is not operating. GDC 60, 61 and 64 read as follows:

> "*Criterion 60–Control of releases of radioactive materials to the environment.* The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.

Criterion 61--Fuel storage and handling and radioactivity control. The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions."

"Criterion 64--Monitoring radioactivity releases. Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents."

Unit 3 was not licensed to the current GDC and has addressed these criteria in FSAR Section 1.3. Unit 3 will continue to meet the requirements of the GDC as follows:

GDC 60 – Compliance will be maintained with requirements to suitably control the release of radioactive materials in gaseous effluents that are produced by normal operation and anticipated operational transients utilizing the FSB emergency ventilation system, the containment, and the containment purge system. These systems continue to retain charcoal and will be tested to assure compliance with 10 CFR 50, Appendix I requirements (the current testing requirements in the TS will be relocated to the FSAR and any modifications that are done under the 10 CFR 50.59 program must assure continued compliance with 10 CFR 50 Appendix I). The proposed changes do not alter the discussion of compliance in FSAR Section 1.3. The criteria of GDC 60 are not applicable to accidents. Nevertheless, accident analyses demonstrate that Containment integrity y is not required to meet acceptable dose criteria.

GDC 61 - Compliance will be maintained with requirements for design of systems that may contain radioactivity (i.e., the FSB emergency ventilation system, containment integrity and the containment purge system). There are no changes being made to these systems other than relocation of the ventilation system testing requirements to the FSAR and the relaxation of integrity requirements. When relocated to the FSAR, all changes to the ventilation systems will require the 10 CFR 50.59 process to be followed which provides assurance of continued compliance with regulatory requirements that are part of the current licensing basis. This is similar to the numerous requirements in FSAR Section 1.3 that specify design requirements the FSB emergency ventilation system must meet. The relaxed containment provisions do not change the containment design features but rather demonstrate through analysis that these feature are not required for fuel handling accidents without recently irradiated fuel.

GDC 64 - Compliance will be maintained with requirements for monitoring the reactor containment atmosphere and FSB atmosphere for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents. The proposed amendment relocates testing requirements for the FSB emergency ventilation system and the containment purge system. It does not change the mode of operation for these systems or monitoring of offsite releases. It simply removes them from TS. The proposed amendment also relaxes TS containment integrity requirements which includes periods where fuel handling accidents without recently irradiated fuel can occur. This creates a potential release path that is unmonitored since containment purge isolation requirements have not been changed. While the amendment relaxes the containment isolation provisions it does not change the plant design which calls for containment purging which will only be isolated if containment release limits are exceeded. In these cases, there are administrative provisions to control the potential for releases by isolating the equipment hatch and personnel air lock(s).

Attachment I – Technical Analysis

5. Although you have supplied an analysis assuming removal of the spray additive tank, the NRC staff's assessment of this amendment will not include a review of that analysis unless you indicate that you are seeking approval of the removal of the spray additive tank.

   Response: Entergy is not seeking approval at this time for the removal of the spray additive tank. This was being considered at the time the calculations were prepared and thus included.

6. It was indicated that for each of the accident scenarios two different control room heating, ventilation, and air conditioning (HVAC) configurations were analyzed. If the intent is to have the option of selecting either configuration in the event of a radiological accident, then it is appropriate to assess both configurations. However, if it is intended that there will only be one configuration, which will be the method of operation for the control room HVAC, then only that configuration will be assessed by the NRC staff. What is the intended modes(s) of operation of the control room HVAC in the event of a radiological accident?

   Response: Entergy has selected the mode with $\geq$1500 CFM filtered make-up and no recirculation for operation in the event of a radiological accident.

7. Will the analysis be amended and submitted to the NRC staff for review and approval if it is determined that the test results of the containment spray system flow rate did not provide adequate margin?

   Response: Yes.

8. The table summarizing the dose limit for the various accidents had incorrect limits for the gas decay tank rupture, the volume control tank rupture and the holdup tank rupture. For AST, the limit should be from 10 CFR Part 20 (i.e., 100 mrem TEDE). This necessitates that these three accidents be re-analyzed to meet the 100 mrem TEDE acceptance criteria or the proposed switch to AST for these three accidents be withdrawn.

   Response: The three identified events have been revised to no longer assume the use of AST. The revised analysis descriptions were provided to NRC staff by Entergy Letter NL-

04-145 as response to NL-04-073 Dos 5. The results of these analyses include thyroid doses, whole body doses, and beta-skin doses (the beta-skin doses are provided only for the control room operators).

Attachment II – Program 5.5.10

9. Items a and b of the program indicate that the in-place acceptance criteria is based upon a penetration of no more than 1%. Has your analysis included a reduction in the effectiveness of the HEPA filters and the charcoal adsorber to account for this 1% penetration?

   Response: The a and b program testing requirements are the in-place testing requirements for the HEPA filters and the charcoal filter bypass and penetration for IP3. For the Control Room Ventilation System and the Containment Fan Cooler Units the analysis assumed a 99% removal efficiency for particulates. For the Control Room Ventilation System the analysis assumed a 90% removal efficiency for elemental and organic (methyl) iodine.

10. Explain why the 1-inch bed of the control room HVAC system is only required to remove 93% of the methyl iodine at a face velocity of 50 feet per minute (ft/min) but must remove 95.5% at a face velocity of 78 ft/min?

   Response: The 1-inch beds are being replaced with 2-inch beds. See the NRC Safety Evaluation, "Technical Specification Amendment for Laboratory Testing of Nuclear-Grade Activated Charcoal Per Generic Letter 99-02," dated October 30, 2003 for a discussion of the removal efficiencies with a one inch and a two inch bed.

Attachment III – Radiological Consequences of Accidents

11. Control room operator doses are provided. For which control room emergency ventilation system operating mode do these doses pertain?

   Response: The doses presented are for the more limiting of the two control room HVAC configurations that were considered and thus bound both of them. Entergy has decided to use the ≥1500 CFM filtered make-up and no recirculation mode.

Loss-of-Coolant Accident (LOCA)

12. At what time following the LOCA was the decontamination factor (DF) of 1000 achieved for particulate? What isotopes was the DF based upon? Provide your calculation that determined when the DF was achieved.

   Response: The DF of 1000 was calculated to occur at 32.91 hours and this value was reduced to 32.8 hours in the dose analysis. The determination of the DF of 1000 was based on a single nuclide, I-131.

   The I-131 releases to the containment atmosphere are modeled as taking place from 30 seconds into the accident until 1.8 hours. The computer code modeled the removal of particulates by the containment sprays starting at 67 seconds. Spray injection phase was assumed to end at 45 minutes with a 3-minute delay before the spray recirculation phase initiated. The recirculation spray effectiveness was assumed to be reduced by half at 3.445 hours (at which time the DF was ~50) and the recirculation sprays were assumed to terminate at 4 hours into the accident. During spray operation, credit was taken for sedimentation removal of particulates in the unsprayed region of the containment and, when the sprays were not active, credit was also taken for sedimentation removal of particulates from the "sprayed" region.

At 4 hours, the particulate I-131 remaining in the containment atmosphere was just over 1.8% of the amount released to the containment for a DF of 55.534. Based on this DF, the additional DF required to reach a final DF of 1000 is 1000 / 55.534 = 18.007.

After 4 hours, the only removal mechanism for particulates is sedimentation with a removal coefficient of 0.1 hr^{-1}. The additional time required to reach a DF of 1000 is calculated by:

$$1/e^{-\lambda t} = DF$$

and: $\ln(1/DF) = -\lambda t$

$$t = \ln(1 / 18.007) / -(0.1) = 28.91 \text{ hr}$$

13. What is the basis for assuming that the airborne fraction of the leakage from the reactor coolant pump is 10%?

    Response: The postulated leakage of sump solution through the RCP seal leak-off line constitutes the total flow into the line. Although this leakage is to a closed system, the CVCS, it has been included in analyses. This results in a slow movement of water through the line with the result that the temperature of the leakage into the Auxiliary Building would be close to ambient. Thus, there would be no flashing of the leaked solution. The assumption of a 10% partition factor for the iodine in the leaked solution is taken from NRC guidance in Appendix A of Regulatory Guide 1.183.

14. What is the basis for assuming that the airborne fraction of emergency core cooling system (ECCS) leakage is 2.7% staring at 6.5 hours after the accident and not a minimum of 10%?

    Response: The value of 2.7% has been replaced by 2.8% (due to rounding but no change in final results). A plant specific calculation has been performed (similar to the calculation performed for Indian Point Unit 2). See Attachments 3 and 4 of this letter for a copy of the proprietary and non-proprietary versions of this calculation.

15. What is the basis for the assumption that there is no sump leakage or reactor coolant pump seal leak-off line leakage between 4 and 6.5 hrs.?

    Response: The assumption of no sump leakage is based on the recirculation system being able to maintain its function under single failure conditions without bringing fluid outside containment. Four hours is allocated to close the valve assumed to be in the open position in the reactor coolant pump seal leak-off line. This is procedurally controlled and the area is accessible as fluid has not yet been brought outside containment for recirculation.

Locked Reactor Coolant Pump Rotor

16. The analyses of the consequences to control room operators should reflect the inleakage characteristics of the control room envelope for the various modes of operation during a radiological accident. Provide the inleakage characteristics of the control room envelope when the normal control room ventilation system is operating and during the time that control room operators are taking manual actions to place the control room emergency filtration system into operation.

    Response: The control room (CR) inleakage characteristics are not currently known since the tracer gas testing has not yet been performed. Tracer gas testing to determine CR inleakage rates is planned for January 2005. The radiological consequences analyses assumed an inleakage rate of 700 cfm during both the normal HVAC operation and when the HVAC is operating in the emergency mode.

Rod Ejection

17. What is the basis for assuming that it will only take 2 hours to stop steam releases from the steam generators and initiate residual heat removal when it takes considerably longer to initiate residual heat removal for other accidents?

    Response: Events which have a essentially intact Reactor Coolant System (e.g., the Steam Generator Tube Rupture and the Main Steam Line Break) would require a substantial period of time before the Residual Heat Removal System can be brought into service. However, with the rod ejection resulting in a breach of the pressure boundary there is a relatively rapid reduction in primary side pressure. As primary side pressure drops below the secondary side pressure, there is a termination of primary-to-secondary leakage and also a termination of heat transfer via the steam generators since there is no recirculation of primary coolant through the steam generators. Thermal-hydraulic analysis shows that the primary side pressure would drop below the secondary side pressure at ~ one hour. In the dose analysis, the time to terminate steaming releases was conservatively extended to two hours.

18. What is the control room envelope inleakage rate during normal operation for this accident?

    Response: See the response to RAI #16.

Small-Break LOCA

19. What is the basis for assuming that it will only take 2 hours to stop steam releases from the steam generators and initiate residual heat removal when it takes considerably longer to initiate residual heat removal for other accidents?

    Response: See the response to RAI #17

Fuel-Handling Accident

20. In Section 11.1.4 it is stated that since credit has not been taken for filtration or containment isolation and that the IP3 analysis supports refueling operations with the

equipment hatch and personnel air lock remaining open. The acceptability of these apertures during fuel handling operations is not limited to obtaining acceptable control room and offsite dose consequences. It is also necessary to demonstrate that the facility remains able to meet GDC 60 and 64. It is also necessary that the removal of such equipment during fuel handling operations meets the criterion of Section II.D of Appendix I to 10 CFR Part 50. Provide additional justification for the equipment hatch and personnel air lock remaining open during fuel handling operations.

Response: NRC has previously approved operation during fuel handling operations with the equipment hatch and personnel air lock remaining open (see NRC Safety Evaluation, "Issuance of Indian Point 3 Amendment 215 for Selective Adoption of Alternate Source Term," dated March 17, 2003).

The request for additional information indicates that the Section 11.1.4, which states that credit has not been taken for filtration or containment isolation and that analysis supports refueling with the equipment hatch and personnel air lock open, fails to address how compliance with 10 CFR 50, Appendix A, GDC 60 and 64 and 10 CFR 50, Appendix I is maintained. The response to question 4 demonstrates continued compliance to GDC 60 and 64. The requirements of Appendix I apply to normal releases and transients so the containment integrity maintained for a fuel handling accident is not a compliance issue. Removal of testing requirements for the containment purge will not affect compliance with the criteria of 10 CFR 50 Appendix I. Entergy has requested removal of the requirements for testing the FSB emergency ventilation system from the TS and will relocate them to the FSAR. Once relocated, the 10 CFR 50.59 process can be used to revise the testing requirements for consistency with 10 CFR 50 Appendix I criteria. A precedent exists for this. The testing requirements for the FSB emergency ventilation system were removed from the Indian Point Unit 2 TS when adopting the alternate source term and relocated to the FSAR. They were subsequently revised to meet 10 CFR 50 Appendix I requirements utilizing the guidance of Regulatory Guide 1.140 insofar as they applied. These requirements insure structural integrity and performance capabilities are maintained, consistent with GDC 61. Because charcoal in the containment purge system is not credited for the fuel handling accident, the requirements of 10 CFR 50.36(c)(2)(ii) no longer require a TS. The requirements of 10 CFR 50.36(c)(2)(ii) are discussed in the response to question 3.

Conclusions

21. Attachment III does not support the complete removal of all filters from the fan coolers if trisodium phosphate baskets are installed as is stated in this section. It was previously stated that an analysis justifying removal of the spray additive was provided for information purposes only and was not part of this amendment request. Since there was no request for the NRC staff to review the analysis, the acceptability of the use of trisodium phosphate baskets in lieu of containment fan coolers cannot be presumed. This section needs to be modified to clarify that this submittal is not a justification for the use of the trisodium phosphate baskets.

    Response: Entergy does not plan to replace the spray additive at this time with trisodium phosphate baskets and is not asking the NRC to review that aspect of the calculations.

The calculations for the large-break LOCA, however, have been performed with NaOH containment spray and without credit for charcoal and HEPA filtration inside containment. Credit for the HEPA filters is included in the calculations for the rod ejection accident and the small-break LOCA.

ATTACHMENT 2 TO NL-04-162

AFFIDAVIT REQUESTING WITHHOLDING POLESTAR CALCULATION

AS PROPRIETARY INFORMATION

(4 PAGES)

ENTERGY NUCLEAR OPERATIONS, INC
INDIAN POINT NUCLEAR GENERATING UNIT 3
DOCKET 50-286

Polestar Applied Technology, Inc.

AFFIDAVIT

I, David E.W. Leaver, being duly sworn, depose and state as follows:

(1) I am a Principal and an Officer of Polestar Applied Technology, Inc. ("Polestar") and am responsible for the function of reviewing the information described in paragraphs (2) and (8) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in portions of Polestar-prepared calculation PSAT 3056CT.QA.04 (see paragraph (8)). This calculation has been prepared for Entergy Nuclear Operations, Inc. in support of an Entergy submittal to NRC on alternate source term (AST) as part of power uprate at Indian Point Unit 3. The Polestar calculation addresses engineered safety feature (ESF) iodine leakage at Indian Point Unit 3.

(3) In making this application for withholding of proprietary information of which it is the owner, Polestar relies upon the exemption from disclosure set forth in the NRC regulations 10 CFR 9.17(a)(4), 2.790(a)(4), and 2.790(d)(1) for "trade secrets and commercial or financial information obtained from a person and privileged or confidential" (Exemption 2.790(a)(4)). The material for which exemption from disclosure is here sought is all "confidential commercial information".

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process or method, including supporting data and analyses, where prevention of its use by Polestar's competitors without license from Polestar constitutes a competitive economic advantage over other companies.

b. Information which, if used by a competitor, would significantly reduce his expenditure of resources or improve his competitive position in the analysis, design, assurance of quality, or licensing of a similar product;

c. Information which reveals cost or price information, production capacities, budget levels, or commercial strategies of Polestar, its customers, or its suppliers;

d. Information which reveals aspects of past, present, or future Polestar customer-funded development plans and programs, of potential commercial value to Polestar;

e. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in both paragraphs (4)a and (4)b, above.

(5) The information sought to be withheld was submitted to Entergy (and, we trust, to NRC) in confidence. The information is of a sort customarily held in confidence by Polestar, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by Polestar, no public disclosure has been made, and it is not available in public sources. All disclosures to third parties including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Distribution of such documents within Polestar is limited to those with a need to know.

(7) The approval of external release of such a document typically requires review by the project manager, and the Polestar Principal closest to the work, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside Polestar are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information identified in paragraph (2), above, is classified as proprietary because it contains detailed information on and results from trade secret methodologies developed by Polestar and applied under the Polestar 10 CFR 50, Appendix B Quality Assurance Program. The trade secret information is identified in [[double bold brackets]] in the calculation. Specifically for ESF leakage calculation PSAT 3056CT.QA.04:

page 3 and pages 5 to 11 of the main calculation body - dealing with Polestar understanding and specially developed methods of realistic modeling of iodine release from ESF leakage pools.

Appendix A, pages 1 - 12 – a detailed calculation of a physical effect which significantly limits the iodine release from pools

Appendix B, pages 2 - 4 – contains details of the actual calculation method for quantifying the iodine release fraction

Attachment 1 (2 pages) and Attachment 2 (4 pages) – contains tables of data and physical parameters which are central to the method for quantifying iodine release fraction

The trade secrets used in this Indian Point 3 work are several of a number of Polestar developed methods, models, and codes. Development of these methods, models, and codes was achieved at a significant cost to Polestar, well over $100,000, which is a significant fraction of internal research and development resources available to a company the size of Polestar.

The development of the methods, models and codes, along with the interpretation and application of the results, is derived from the extensive experience database that constitutes a major Polestar asset.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to Polestar's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of Polestar's comprehensive technology base on application of the AST to operating plants and advanced light water reactors, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with methods which have been developed and are being maintained in accordance with 10 CFR 50, Appendix B requirements.

The research, development, engineering, analytical and review costs comprise a substantial investment of time and money by Polestar.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

Polestar's competitive advantage will be lost if its competitors are able to use the results of the Polestar experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to Polestar would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive Polestar of the opportunity to exercise its competitive advantage to seek an adequate return on its relatively large investment in developing these very valuable analytical tools.

STATE OF CALIFORNIA)
) ss:
COUNTY OF SANTA CLARA)

David E.W. Leaver, is being duly sworn, deposes and says:

That he has read the foregoing affidavit and the matters stated therein are true and correct to the best of his knowledge, information, and belief.

Executed at Los Altos, California, this 5th day of November 2004.

David E W Leaver

David E.W. Leaver
Polestar Applied Technology, Inc.

Subscribed and sworn before me this 5th day of November 2004.

RUTH TUBBS
Commission # 1439944
Notary Public - California
Santa Clara County
My Comm. Expires Oct 8, 2007

Ruth G Tubbs

Notary Public, State of California