April 11, 2000

Mr. J. H. Swailes
Vice President of Nuclear Energy
Nebraska Public Power District
P. O. Box 98
Brownville, NE 68321

SUBJECT: COOPER NUCLEAR STATION - ISSUANCE OF AMENDMENT ON AVERAGE POWER RANGE MONITOR NEUTRON FLUX-HIGH (FLOW-BIASED) ALLOWABLE VALUE (TAC NO. MA7705)

Dear Mr. Swailes:

The Commission has issued the enclosed Amendment No. 184 to Facility Operating License No. DPR-46 for the Cooper Nuclear Station (CNS). The amendment consists of changes to the Technical Specifications (TS) in response to your application dated December 15, 1999, as supplemented by letters dated February 15 and April 8, 2000.

The amendment revises the CNS TS regarding the Average Power Range Monitor (APRM) Neutron Flux-High (Flow-Biased) allowable value. This TS change allows CNS to operate in the Maximum Extended Load Line Limit (MELLL) domain, along a 121-percent rod line, and adds the Increased Core Flow (ICF) up to 105-percent flow domain.

Proper planning regarding submission of licensing requests is necessary to allow sufficient allocation of NRC and Nebraska Public Power District (NPPD) resources. NPPD submitted three amendment requests in December 1999 and requested an issuance date that would support CNS startup in early April 2000. In light of the scope and complexity of the requested changes, and the importance of the issues, the staff feels that sufficient preparation and planning was not given to these issues by NPPD.

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

Sincerely,

/RA/

Lawrence J. Burkhart, Project Manager, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-298

Enclosures: 1. Amendment No. 184 to DPR-46
2. Safety Evaluation

cc w/encls: See next page

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Cooper Nuclear Station

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NEBRASKA PUBLIC POWER DISTRICT

DOCKET NO. 50-298

COOPER NUCLEAR STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 184
License No. DPR-46

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

   A. The application for amendment by Nebraska Public Power District (the licensee) dated December 15, 1999, as supplemented February 15 and April 8, 2000, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of Facility Operating License No. DPR-46 is hereby amended to read as follows:

   2. Technical Specifications

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 184, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of its date of issuance and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Robert A. Gramm, Chief, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: April 11, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 184

FACILITY OPERATING LICENSE NO. DPR-46

DOCKET NO. 50-298

Replace the following pages of the Appendix A Technical Specifications with the enclosed revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
3.3-6	3.3-6

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 184 TO FACILITY OPERATING LICENSE NO. DPR-46

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

1.0 INTRODUCTION

By letter dated December 15, 1999 (Ref. 1), Nebraska Public Power District (NPPD) submitted proposed changes to the Cooper Nuclear Station (CNS) Technical Specifications (TS). The TS change request would revise the Average Power Range Monitor (APRM) Neutron Flux-High (Flow-Biased) allowable value to allow incorporation of the Maximum Extended Load Line Limit (MELLL) and Increased Core Flow (ICF) regions for CNS operation.

On February 15, 2000, the licensee submitted by letter (Ref. 2) a General Electric Nuclear Energy (GENE) safety evaluation to support the request, GENE report NEDC-32914P, "Maximum Extended Load Line Limit and Increased Core Flow for Cooper Nuclear Station," (Ref. 3). The Maximum Extended Load Line Limit (MELLL) operation mode and the associated TS changes expand the operating domain along the 121 percent rod line to the power/flow point of 100 percent power and 75 percent core flow. The ICF domain extends the core flow to 105 percent of rated flow up to rated power. The current Power to Flow Map is based on a region bounded by the Extended Load Line Limit (ELLL) and evaluations that are prepared as part of the Core Operating Limits Report.

In a letter dated April 8, 2000, NPPD stated that the calculated peak drywell temperature (see containment response evaluation in section 3.4) resulting from operation in the MELLL was determined to be acceptable as containment structural loads remained within limits and because the peak temperature was bounded by the maximum allowable drywell airspace temperature of 309 ^{0}F, contained in CNS TS Bases Section B 3.6.1.5. As noted in the letter, upon further investigation, NPPD determined that the value of 309 ^{0}F may require revision and will be evaluated within its Corrective Action Program. NPPD stated that the airspace acceptance criteria may change, however, this change does not impact the acceptability of the calculated peak drywell airspace temperature submitted in the February 15, 2000, letter. NPPD stated that the revised acceptance criteria is at least equal to the calculated peak drywell temperature contained in the February 15, 2000, letter.

The February 15 and April 8, 2000, letters provided additional clarifying information that was within the scope of the original application and *Federal Register* notice and did not change the staff's initial proposed no significant hazards consideration determination.

2.0 BACKGROUND

The APRM Neutron Flux-High (Flow Biased) Function monitors neutron flux to approximate the thermal power being transferred to the reactor coolant. The APRM neutron flux trip level is varied as a function of recirculation drive flow (i.e., at lower core flows, the setpoint is reduced proportional to the reduction in power experienced as core flow is reduced with a fixed control rod pattern) but is clamped at an upper limit, the Function 2.c, APRM Neutron Flux-High (Fixed) Function Allowable Value. The APRM Neutron Flux-High (Flow-Biased) Function is not specifically credited in the safety analyses, but is intended to provide protection against transients where thermal power increases slowly, and to provide protection for power oscillations which may result from reactor thermal hydraulic instability.

The APRM Neutron Flux-High (Flow Biased) setpoints can restrict operational flexibility of the plant in achieving and maintaining rated power. The allowable value places an upper limit on the setpoints and thus may also restrict operational flexibility. With the expansion of the Power to Flow Map to the region bounded by the MELLL, the effect of these restrictions on operation becomes more severe. The operating restriction can be relaxed with the implementation of an increased flow biased allowable value, which requires a TS amendment. The proposed change does not relax operating restrictions in the ICF region as the flow biased allowable value is limited by the 119 percent reactor thermal power fixed value at high core flows. No change to the 119 percent fixed value is proposed.

Operation in the expanded Power to Flow Map regions enhances the ability to more efficiently approach and maintain operation at rated power by allowing operation to be maintained with recirculation flow over a wider flow range. In addition, less frequent rod adjustments would be required to compensate for reactivity depletion. The need to perform power reductions to perform control rod withdrawals would also be decreased. As a result, the plant would be able to operate for longer periods at rated power, have more flexibility to schedule load reductions, and be able to operate in a safer, more efficient, and more economical manner.

In order to support evaluation of operation within the MELLL and ICF regions, which is facilitated by this TS change, NPPD submitted NEDC-32914P, “Maximum Extended Load Line Limit and Increased Core Flow for Cooper Nuclear Station,” dated January, 2000. Impacts on containment, the reactor vessel, recirculation system, reactor vessel internals, limiting transients for Cycle 20, loss-of-coolant accident (LOCA), and anticipated transients without scram (ATWS) events were evaluated.

3.0 EVALUATION

The staff has reviewed the GENE evaluation of the impacts of CNS operation in the MELLL and ICF region (Ref. 3). The evaluation addresses impacts on anticipated operational occurrence (AOO) events, vessel overpressure protection analysis, LOCA analysis, containment response, reactor internal pressure differences, reactor internals structural integrity, reactor internal vibration, reactor recirculation system evaluation, thermal hydraulic stability, and ATWS. CNS operation in the MELLL and ICF regions could potentially impact these analyses due to allowed operation in the new revised power to flow map.

Both the core-wide AOOs and the vessel overpressure protection safety analyses were performed for CNS Cycle 20, which includes the first reload batch of the new GE14 fuel design.

The emergency core cooling system analyses were performed for the limiting GE14 fuel design. The other analyses in the report are independent of the fuel cycle.

3.1 Anticipated Operational Occurrences (AOOs) for CNS Operation in the MELLL and ICF Regions

The limiting AOOs (pressurization and non-pressurization events) for operation in the MELLL and ICF domains were examined for CNS Cycle 20 operation to ensure that operating limit minimum critical power ratio (OLMCPR) requirements are satisfied. Maintaining the OLMCPR provides sufficient margin so that the safety limit MCPR (SLMCPR) will not be exceeded. Maintaining this limit assures that 99.9 percent of the fuel rods are expected to avoid boiling transition.

The following core-wide events were considered to bound OLMCPR requirements and were evaluated for MELLL and ICF operation:

- Generator Load Rejection with No Bypass (LRNBP)
- Turbine Trip with No Bypass (TTNBP)
- Feedwater Controller Failure Maximum Demand (FWCF)
- Loss of 100 °F Feedwater Heater (LFWH)

The transient and accident analysis methodologies used for the CNS Cycle 20 safety evaluation are described in GESTAR-II (Ref. 4). The analytical methods, as well as the input assumptions, such as reactor protection system setpoints and plant configurations, are consistent with the reload analysis. The core-wide rapid pressurization events (LRNBP, TTNBP, and FWCF) and the LFWH events are limiting for these two operating domains because the other potentially limiting events, such as mis-located bundle and rotated bundle, were analyzed and determined to be bounded by these events. These events were examined for OLMCPR impact when operating in the MELLL and ICF regions. The results of the evaluation showed that the MCPR operating limits for 100 percent power/100 percent flow and MELLL are bounded by the ICF operating condition and that the LRNBP is the limiting AOO event (thus, future reload analyses will be bounded by the 100 percent/105 percent flow initial condition). The results of the analysis are that OLMCPRs are maintained above the MCPR safety limit. Therefore, the analysis and results are acceptable.

3.2 Vessel Overpressure Protection Analysis

The main steam isolation valve (MSIV) closure with a flux scram event was analyzed and demonstrated to conform to the American Society of Mechanical Engineers (ASME) Pressure Vessel Code at ICF conditions for Cycle 20 at 102 percent power and 105 percent core flow. The results of this analysis show the peak pressure is 1246 psig with eight safety relief valves and three system safety valves set 3 percent above their nominal value. This pressure is well within the ASME Section III limit of 1375 psig. Therefore, the analysis and results are acceptable.

3.3 Loss-of-Coolant Accident (LOCA) Analysis

A LOCA analysis was conducted to determine the impact of MELLL and ICF operation on the CNS LOCA peak cladding temperature (PCT). The result with the limiting GE14 fuel design at the MELLL condition was an estimated increase of 50 ^{0}F from the GE8x8NB fuel design in the Cycle 20 reload core at rated core flow. The analysis of the GE14 fuel design at the ICF condition resulted in no change in PCT compared to rated core flow. The evaluation was conducted using the approved GESTAR-II methodology SAFER/GESTAR-LOCA. The GE14 Licensing Basis PCT of 1760 ^{0}F was calculated at the MELLL point which is well below the 10 CFR 50.46, 2200 ^{0}F limit. Therefore, the analysis and results are acceptable.

3.4 Containment Response

Bounding short-term containment response analyses of the design-basis LOCA event were performed to demonstrate that operation in the MELLL and ICF domains will not result in exceeding containment design limit. The CNS updated safety analysis report short-term containment evaluation is dependent on reactor initial pressure and temperature that change with MELLL and ICF operation. The long-term heatup of the suppression pool following a LOCA is governed by the decay heat which depends on the reactor rated power level which remains unchanged with either MELLL or ICF operation.

For the short-term containment response of the design-basis LOCA, the maximum peak drywell pressure is 54.4 psig at 102 percent power and 75 percent core flow, which is below the design value of 56 psig. The maximum peak wetwell pressure is 24.0 psig, which is well below the design value of 56 psig. Therefore, this value is acceptable to the staff.

The maximum peak drywell temperature is 301.4 ^{0}F at the point of maximum vessel subcooling (62 percent power/34 percent flow), which is below the maximum allowable drywell airspace temperature acceptance criterion. In its letter dated February 15, 2000, NPPD stated that this acceptance criterion was 309 ^{0}F, which is contained in CNS TS Bases Section B 3.6.1.5. In its letter dated April 8, 2000, NPPD stated that this acceptance criterion may require revision and is being evaluated within NPPD's Corrective Action Program. However, NPPD stated that this does not impact the acceptability of the calculated peak drywell airspace temperature because the revised acceptance criteria will be at least equal to 301.4 ^{0}F. Based on the calculated maximum peak drywell temperature being less than or equal to the maximum allowable drywell airspace temperature acceptance criterion, the staff finds this acceptable.

3.5 Reactor Vessel Internals Pressure Differences and Structural Integrity

Operation in either the MELLL or ICF domain affects the pressure differences across reactor internal components. Operation in the ICF region (up to 105 percent rated flow) results in higher initial flow velocities and causes increased reactor internal pressure differences (RIPD) across the internal components for Normal, transient (Upset), Emergency, and accident (Faulted) conditions. The impact of MELLL operation on RIPD is bounded by ICF operation.

Analyses of Normal operating conditions were performed with the steady-state thermal-hydraulic model at 100 percent power/105 percent flow. The inputs used for this analysis are consistent with the original CNS RIPD with the assumption of a full core of the limiting fuel (GE14) for pressure drop consideration.

For Upset conditions, the steady-state (Normal condition) values are conservatively adjusted to obtain the limiting AOO RIPDs. However, the initial steady-state pressure differences at the low flow conditions (MELLL and ELLL) would be smaller than for ICF at the same power level because of the lower initial flow velocity. Consequently, it is bounding to apply Upset conditions adjustment factors to the conservative ICF steady-state results.

Emergency RIPDs were obtained by using the LAMB model to analyze the limiting All Automatic Depressurization System (ADS) Valves Actuation Event.

Faulted RIPD values are obtained using the LAMB computer code to analyze the limiting steamline break accident. No MELLL specific calculation is required because MELLL is bounded by ICF. The Faulted condition RIPD calculation for CNS also includes an evaluation at the low power cavitation interlock point (22.5 percent power/110 percent flow).

These methods are acceptable to the staff to determine pressure drops across the internal components with ICF operation and were used to perform evaluations of the reactor internal structural integrity. The structural adequacy of the internal components was assessed for load changes associated with MELLL and ICF operation, using the original/existing analysis as the design basis. All loads and stresses were within the design-basis allowable values for each of the components reviewed. The staff considers the analyses and results to be acceptable and concludes that CNS can operate in the MELLL and ICF regions without any detrimental effects on the reactor internals due to reactor internal pressure differences.

3.6 Reactor Internal Vibration

Evaluations of changes in the flow-induced vibration response of critical components within the reactor pressure vessel due to operation in the MELLL and ICF modes were performed. All safety-related reactor internal components had vibration stresses less than the acceptance criteria at ICF conditions, except for two jet pump sensing lines (one per recirculation loop) that were found to have a remote possibility of a second natural frequency near the maximum recirculation pump speed, only if five as-built lengths were at their extreme design value tolerances. Although it is highly improbable that these two sensing lines actually have natural frequencies near the maximum pump speed, a single sensing line in a jet pump pair is sufficient to detect any jet pump anomaly. Therefore failure of one jet pump would not affect the ability to detect vibration as recirculation pump speed is increased into the ICF region. Therefore, the analysis and results are acceptable.

3.7 Reactor Recirculation System Evaluation

The reactor recirculation system (RRS) provides forced circulation of coolant water through the reactor core and the external RRS piping. The maximum core flow rate is limited by the total hydraulic resistance inside the reactor vessel. The maximum RRS performance can vary from cycle to cycle due to core pressure drop and jet pump efficiency. An evaluation was performed based on original equipment design conditions with core hydraulic conditions equivalent to Cycle 19 and Cycle 20 operation. The staff has determined that the 5-percent increase in core flow (ICF) is supported by the original RRS design.

3.8 Thermal Hydraulic Stability

CNS has currently implemented reactor stability Long-Term Solution Option I-D, which provides an administratively controlled exclusion region in the power/flow map to prevent normal operation where an instability could be expected to occur. There is also a buffer zone outside the exclusion region of 5 percent of rated power and rated core flow. A cycle-specific reload analysis was used to determine the boundaries of the exclusion region. The analysis demonstrated that SLMCPR protection is provided on the rated flow control line by the APRM Neutron Flux High (Flow Biased) setpoints which are bounded by the revised allowable value as described in Section 3.10. The Cycle 20 reload core design, including a mixed core of current GE9 and new GE14 fuel designs was used. The endpoints of the exclusion region are defined on the MELLL line and on the natural circulation line. The endpoints of the buffer zone are defined as 5 percent of rated flow higher along the MELLL line and 5 percent of rated power lower along the natural circulation line. The CNS Cycle 20 OLMCPR value of 1.38 was found to be lower (less limiting) than the limiting plant OLMCPR of 1.46. Since the stability OLMCPR is not limiting for Cycle 20 and the stability exclusion region was reanalyzed for new fuel and reload core designs to provide assurance of stability performance, the staff finds the analysis and results acceptable.

3.9 Anticipated Transients Without Scram (ATWS)

The regulatory requirements for postulated ATWS events are contained in 10 CFR 50.62. For boiling-water reactors (BWRs), the rule includes requirements for an automatic ATWS recirculation pump trip, an alternate rod insertion system, and an 86-gallon-per-minute equivalent standby liquid control system. The following criteria demonstrate that these requirements are met for plant changes which can affect response to ATWS events:

- Reactor vessel integrity is maintained (peak vessel pressure is less than ASME Service Level C limit)

- Containment integrity is maintained (peak suppression pool temperature is less than peak suppression pool temperature limit for containment analysis and peak containment pressure is less than containment design pressure)

- Fuel integrity is maintained (PCT and peak cladding oxidation are below the corresponding 10 CFR 50.46 limits)

A plant-specific ATWS analysis was performed to support CNS operation in the MELLL domain with a full-core loading of the new GE14 fuel. Four events were analyzed based on potential challenges to the above criteria:

- Closure of all MSIVs

- Pressure regulator failure to maximum demand

- Loss of auxiliary power, and

- Inadvertent opening of one safety relief valve.

Conformance to the ATWS acceptance criteria of 10 CFR 50.62 was maintained for each of these four limiting events for operation in the MELLL region. Specifically, (1) the PCT and cladding oxidation are below the limits specified in 10 CFR 50.46, (2) the peak vessel pressure is below the ASME Service Level C limit of 1500 pounds per square inch and meets the ATWS overpressure criteria, and (3) the peak suppression pool temperature is below the maximum containment temperature limit and below the long-term maximum suppression pool temperature calculated for LOCA conditions and the peak containment pressure is well below the containment design pressure. Therefore, this analysis and results are acceptable to the staff.

3.10 Proposed CNS TS APRM Neutron Flux-High (Flow Biased) Allowable Value Changes

The proposed TS change is to revise the APRM Neutron Flux-High (Flow Biased) Allowable Value as defined in Function 2.b on Table 3.3.1.1-1 of TS 3.3.1.1, “Reactor Protection System Instrumentation.” This is requested to facilitate operation in the MELLL region (the proposed change does not relax operating restrictions in the ICF region as the flow biased allowable value is limited by the 119 percent reactor thermal power fixed value at high core flows and no change to the 119 percent fixed value is proposed).

For the current licensed ELLL power/flow map, the flow-biased APRM neutron flux-high (flow biased) allowable value was defined as "$\leq$0.58 W+ 61.0 percent RTP and $\leq$119 percent RTP[(b)]," where W is the recirculation drive flow in percent of rated and RTP is the reactor thermal power. The proposed change to facilitate operation in the MELLL region is for the allowable value to be defined as "$\leq$0.66 W + 71.5 percent RTP and $\leq$119.0 percent RTP[(b)]." There is no proposed change to the upper limit of 119 percent RTP for operation in the ICF region.

Additionally, for single-loop operation footnote (b) to TS Table 3.3.1.1-1 will be changed <u>from</u> "0.58 W + 61.0 percent - 0.58 ΔW RTP" <u>to</u> "0.66 W + 71.5 percent - 0.66 ΔW RTP" when reset for single-loop operation per Limiting Condition for Operation 3.4.1, 'Recirculation Loops Operating' to match the new MELLL settings. ΔW is defined as the difference between two loop and single loop effective drive flow at the same core flow (e.g., ΔW equals zero for two recirculation loop flow).

Changes to the allowable value will enable changes to the setpoints. The licensee has stated that the establishment of the revised setpoints will be consistent with the previously approved GE setpoint methodology, will be based on the analytical limit supported by reference 3, and will be implemented in accordance with plant procedures.

As stated in TS Bases 3.3.1.1, the APRM Neutron Flux-High (Flow Biased) Function is not specifically credited in the safety analyses, but is intended to provide protection against transients where thermal power increases slowly, and to provide protection for power oscillations which may result from reactor thermal hydraulic instability (see above evaluations).

Due to the acceptability of NPPD's evaluation of CNS operation in the MELLL and ICF regions as described above and the fact that the APRM Neutron Flux High (Flow Biased) allowable value is not credited in any safety analysis, the changes to the CNS TS as described above are acceptable to the staff. In addition, NPPD has stated that the establishment of the revised setpoints is consistent with the previously approved GE setpoint methodolgy.

4.0 <u>STATE CONSULTATION</u>

In accordance with the Commission's regulations, the Nebraska State official was notified of the proposed issuance of the amendment. The State official had no comment.

5.0 <u>ENVIRONMENTAL CONSIDERATION</u>

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (65 FR 4279). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Section 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

1. Letter from J. Swailes to USNRC, “Proposed Changes to CNS Technical Specifications - Change to Average Power Range Monitors (APRMs) Neutron Flux-High (Flow Biased) Allowable Value,” dated December 15, 1999.

2. Letter from J. Swailes to USNRC, Maximum Extended Load Line Limit Analysis (MELLLA) - Submittal of Final Report,” dated February 15, 2000.

3. NEDC-32914P, “Maximum Extended Load Line Limit and Increased Core Flow for Cooper Nuclear Station,” dated January, 2000. (Proprietary information. Not publicly available. A nonproprietary version is available.)

4. NEDE-24011-P-A-11, “General Electric Standard Application for Reactor Fuel,” GESTAR II, Revision 13, August 20, 1996. (Proprietary information. Not publicly available.)

Principal Contributor: E. Kendrick

Date: April 11, 2000