containment purge valves and the supplementary containment purge valves (TSs 4.6.1.7.2 and 4.6.1.7.3).

The NRC staff has denied the portion of the proposed change regarding the frequency of leakage rate testing the normal containment purge valves and the supplementary containment purge valves. These valves use resilient seals. The licensee proposed to extend the present test intervals of 3 months for the supplementary purge valves and 6 months for the normal purge valves following the guidance of RG 1.163. RG 1.163 recommends testing of containment purge and vent valves at intervals not exceeding 30 months. However, the current test intervals are not based on Appendix J considerations and the licensee's proposal is therefore outside the scope of the proposed change to Option B. The current test intervals are based on the findings of Generic Issue B–20, ''Containment Leakage Due to Seal Degradation,'' that valves with resilient seals should be tested more frequently than required by Appendix J. The background for this conclusion is discussed in IE Circular 77–11, ''Leakage of Containment Isolation Valves With Resilient Seats,'' issued on September 6, 1977.

After some discussions with the staff, the licensee chose not to pursue this issue further. Since additional information would be required to continue this part of the review (for TSs 4.6.1.7.2 and 4.6.1.7.3), the staff denies this part of the proposed change.

The licensee was notified of the Commission's denial of the proposed change by a letter transmitting Amendment Nos. 84 and 71.

By September 19, 1996, the licensee may demand a hearing with respect to the denial described above. Any person whose interest may be affected by this proceeding may file a written petition for leave to intervene.

A request for hearing or petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, by the above date. A copy of any petitions should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and to Jack R. Newman, Esq., Morgan, Lewis & Bockius, 1800 M Street, N.W., Washington, D.C. 20036–5869, attorney for the licensee.

For further details with respect to this action, see (1) the application for amendment dated May 1, 1996, and (2) the Commission's letter to the licensee dated August 13, 1996, issued with Amendment Nos. 84 and 71 to NPF–76 and NPF–80.

These documents are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room located at the Wharton County Junior College, J.M. Hodges Learning Center, 911 Boling Highway, Wharton, TX 77488.

Dated at Rockville, Maryland, this 13th day of August, 1996.

For the Nuclear Regulatory Commission.

Thomas W. Alexion,

Project Manager, Project Directorate IV–1, Division of Reactor Projects III/IV, Office of Nuclear Reactor Regulation.

[FR Doc. 96–21164 Filed 8–19–96; 8:45 am]

BILLING CODE 7590–01–P

[Docket Nos. 50–245, 50–336, and 50–423; License Nos. DPR–21, DPR–65, and NPF–49]

Northeast Nuclear Energy Company, (Millstone Nuclear Power Station Units 1, 2, and 3); Confirmatory Order Establishing Independent Corrective Action Verification Program (Effective Immediately)

I

Northeast Nuclear Energy Company (Licensee) is the holder of Facility Operating License Nos. DPR–21, DPR–65, and NPF–49 issued by the Nuclear Regulatory Commission (NRC or Commission) pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50 on October 31, 1986,[1] September 26, 1975, and January 31, 1986 respectively. The licenses authorize the operation of Millstone Units 1, 2 and 3 in accordance with conditions specified therein. All three facilities are located on the Licensee's site in Waterford, Connecticut.

II

On August 21, 1995, as supplemented August 28, 1995, the NRC received a petition under 10 CFR 2.206 which requested that NRC shut down Millstone Unit 1 and take enforcement action based upon alleged violations of NRC requirements related to operation of the spent fuel pool cooling systems and refueling practices. On November 4, 1995, the Licensee shut down Millstone Unit 1 for a planned 50-day refueling outage. During the fall of 1995, an NRC investigation of licensed activities at Millstone Unit 1 identified potential violations regarding refueling practices and the operation of the spent fuel pool cooling systems of Millstone Unit 1. On December 13, 1995, the NRC issued a letter to the Licensee requiring that it inform the NRC, pursuant to Section 182a of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f), with regard to Millstone Unit 1, of the actions it would be taking to ensure that future operation of that facility would be conducted in accordance with the terms and conditions of the plant's operating license, the Commission's regulations, including 10 CFR 50.59, and the plant's Updated Final Safety Analysis Report (UFSAR).

On February 20, 1996, the Licensee shut down Millstone Unit 2 when both trains of the high pressure safety injection (HPSI) system were declared inoperable due to the potential to clog the HPSI discharge throttle valves during the recirculation phase following a loss-of-coolant accident (LOCA). On February 22, 1996, the Licensee issued Adverse Condition Report (ACR) 7007—Event Response Team Report, which describes in detail the underlying causes for numerous inaccuracies contained in Millstone Unit 1's UFSAR. Those causes, as determined by the Licensee, include the following: (1) Errors and omissions in the original 1986/87 UFSAR; (2) failure of the administrative control programs to address fully NRC requirements; (3) failure of the Licensee to implement fully those administrative programs; (4) a pattern of failure of Licensee management to correct identified weaknesses and risks associated with the UFSAR and design bases; and (5) failure of Licensee oversight to identify this pattern to management, the significance of the pattern itself, or the ineffectiveness of corrective actions to prevent its recurrence. The report acknowledged that, due to the nature of these identified causes, the potential existed for the presence of similar configuration management problems at Connecticut Yankee and Millstone Units 2 and 3.

In response to the Licensee's ACR 7007 and the NRC's own ongoing inspections, evaluations and investigations, on March 7, 1996, the NRC issued a letter to the Licensee requiring that it inform the NRC, pursuant to Section 182a of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f), with regard to Millstone Unit 2, of the actions it would be taking to ensure that future operation of that facility would be conducted in

[1] Millstone Unit 1 was issued its provisional operating license on October 7, 1970 and commenced operation on March 1, 1971. This unit received a full term operating license on October 31, 1986.

accordance with the terms and conditions of the plant's operating license, the Commission's regulations, including 10 CFR 50.59, and the plant's UFSAR. The letter stated that this information was to be submitted no later than 7 days prior to the Unit's restart (prior to criticality) from its current outage. The Millstone Unit 2 letter also described findings the NRC had made in recent inspections of that facility which suggested that significant operability and design concerns remained, including the HPSI issue identified above, as well as inadequate containment sump screen mesh and a flawed post-accident containment hydrogen monitor design.

On March 7, 1996, the NRC also issued a 50.54(f) letter to the Licensee regarding the Millstone Unit 3 plant, which was then operating at full power. In that letter, the NRC noted that it did not have an inspection history at Millstone Unit 3 that revealed design deficiencies similar in number and nature to that of Millstone Units 1 and 2. Nonetheless, the NRC concluded that it required additional information, within 30 days of the date of the letter, including the Licensee's plans and actions to address the implications of ACR 7007 for Millstone Unit 3, as well as the Licensee's plans and schedules to ensure that future operation of the unit would be conducted in accordance with the Commission's regulations, the terms and conditions of the operating license, and the facility UFSAR.

Following the March 7 letter, the NRC conducted a special inspection at Millstone Unit 3 that identified design and other deficiencies similar to those reported in ACR 7007 and by the NRC at the other Millstone units. On March 30, 1996, Unit 3 was shut down after it was determined that containment isolation valves for the auxiliary feedwater (AFW) turbine-driven pump were inoperable due to the valves' noncompliance with NRC requirements. Shortly thereafter, while still shut down, the Licensee discovered that the facility had been operating in a condition outside its design basis due to the Licensee's failure to adequately address design temperature conditions in the stress calculations for the Containment Recirculation Spray System (RSS) piping and supports. Both of these deficiencies had existed for over ten years, since initial operation of the facility. All three Millstone Units remain shut down.

On April 4, 1996, the NRC issued a second letter to the Licensee, pursuant to 10 CFR 50.54(f), with regard to Millstone Unit 3, similar to those issued for Millstone Units 1 and 2. The letter described programmatic issues and design deficiencies identified during the NRC's ongoing special inspection of the plant that were similar in nature to those present at Millstone Units 1 and 2. These included the inoperability of the turbine-driven AFW pump during startup and shutdown, the failure to remove plastic shipping plugs from Rosemount transmitters, the failure to correct a degraded non-safety battery, inadequate control of the modification of the service water system, and the potential for introduction of foreign material into the containment sump. In addition, the letter noted Licensee-identified design deficiencies in the AFW containment isolation valves and RSS that had existed for more than 10 years. As in the case of the Millstone Unit 1 and 2 letters, as described above, the Licensee was required to provide the NRC, no later than 7 days prior to the Unit's restart, with information necessary to assure the NRC that the plant will be operated in conformance with the terms and conditions of the plant's operating license, the Commission's regulations, including 10 CFR 50.59, and the plant's UFSAR.

On May 21, 1996, pursuant to 10 CFR 50.54(f), the NRC issued a letter to the Licensee requiring specific information regarding design and configuration deficiencies identified at each of the Millstone units as well as a detailed description of the Licensee's plans for completion of the work required to respond to the NRC's letters of December 13, 1995, March 7, 1996, and April 4, 1996. The NRC required this information to be submitted within 30 days of the date of the letter for the first unit that the Licensee proposed to restart and not later than 60 days prior to the Licensee's proposed restart for the remaining Millstone units.

Based upon the Licensee's assessment of the extent and scope of identified design control problems at Millstone Station, the Licensee decided to focus its near-term efforts on restart of Millstone Unit 3. In a letter dated June 20, 1996, the Licensee responded to the NRC's May 21, 1996, letter and informed NRC that Millstone Unit 3 would be the first Millstone unit the Licensee proposed to restart. In Attachment 1 to its June 20 response, the Licensee listed 881 design and configuration deficiencies identified since issuance of ACR 7007 and entered into the Licensee's Deficiency Review Team Report database as of June 13, 1996. The Licensee designated 378 items to be corrected prior to restart of Millstone Unit 3. The Licensee determined that the items it had designated for correction prior to restart, if not corrected, could impact upon operability of required equipment, raise unreviewed safety questions, or indicate discrepancies between the plant's UFSAR and the as-built plant or operating procedures.

In the June 20 letter, the Licensee also described its own Configuration Management Plan (CMP), intended to provide reasonable assurance that the future operation of Millstone Units 1, 2, and 3 will be conducted in accordance with the terms and conditions of their applicable operating licenses, UFSARs and NRC regulations. The CMP includes efforts to understand licensing and design basis issues which led to issuance of the 50.54(f) letters and actions to prevent those issues' recurrence. Additionally, the Licensee described its CMP objective to clearly document and meet the units' licensing and design basis requirements, and its intention to ensure that adequate programs and processes exist to maintain control of licensing and design basis requirements.

On July 2, 1996, the Licensee supplemented its June 20, 1996 response to NRC's May 21, 1996 50.54(f) letter. The Licensee provided additional information on Millstone Unit 3 deficiencies previously reported, identified revisions to its plans and committed to complete a review to identify and correct, as necessary, Millstone Unit 3 UFSAR deficiencies prior to restart. The Licensee reported a substantial increase in the total number of identified design and configuration management discrepancies (1187 items), and an increase in those proposed by the Licensee for corrective action prior to restart (597 items).

As the Licensee's own submissions and NRC inspections indicate, significant design control deficiencies and degraded and non-conforming conditions have been identified at Millstone Units 1, 2, and 3. The staff has identified three major types of design control problems which exist at all three Millstone plants. Specific examples of deficiencies at each plant in each of the categories are provided below.

1. Errors in Licensing/Design Basis Documentation

The NRC identified errors in the UFSARs for Millstone Units 1, 2, and 3. For example, at Millstone Unit 3, the protective relay settings and calculations for 4kv safety-related motor feeders were not set consistent with the UFSAR. At Millstone Unit 2, the UFSAR indicated that certain non-essential loads of the reactor building closed cooling water (RBCCW) system inside containment were automatically isolated during a sump recirculation actuation signal when in fact the associated isolation valves received no automatic isolation signal. Additionally, the RBCCW flow rates assumed in the accident analyses were non-conservative with respect to the actual system flow rates.

In addition, the NRC found instances of modifications that were completed without implementing required revisions to the UFSAR. For example, the Licensee revised the Millstone Unit 3 Technical Specifications (TS) in January 1995 to change the testing frequency of the auxiliary feed pumps from monthly to quarterly, but did not update the UFSAR to reflect the change.

At Unit 1, the Licensee failed to perform and document a safety evaluation for an electrical separation deficiency associated with a feedwater regulating valve interlock. This deficiency was not corrected and constituted a change to the design of the facility as described in the UFSAR. Also, the Licensee's assessment of the need for upgrades to the intake structure ventilation system was inadequate. Specifically, insufficient heat removal capability existed under several postulated scenarios.

At Unit 2, the NRC found that the UFSAR had not been updated to reflect that the intake structure design temperature could not be met following a loss of non-vital exhaust fans.

Furthermore, while the Millstone Unit 3 UFSAR documented that the design bases for the containment heat removal systems had been established in accordance with specific general design and code criteria, portions of these systems were found to violate certain analytical stress considerations. Specifically, the recirculation spray system (RSS) pipe supports inside containment were not designed to withstand a single failure of a supporting service water train. Also, both the RSS and quench spray systems were found to contain pipe supports for which ASME Code stress allowables would be exceeded during design basis accident temperature conditions within the Unit 3 containment building.

2. Failure To Translate Design Bases to Procedures and Hardware

The NRC found instances where the Licensee did not adequately translate design basis information into procedures, practices, hardware and drawings. For example, at Millstone Unit 1, the reactor pressure assumed as an initial condition in the accident analyses was exceeded during reactor power operation. At Unit 3, a modification that installed the service water intake structure sump pump called for specific periodic testing, but such testing was never performed. In another case at Unit 3, prelubrication of the AFW pump was not performed every 40 days as required by the vendor.

As noted in the NRC's letter of December 13, 1995, at Millstone Unit 1, the Licensee's core offload practices were not consistent with the Unit's UFSAR. Specifically the heat load assumptions were not maintained as a result of full core offloads performed sooner than the required delay time after reactor shutdown.

Also at Unit 1, measures established to ensure that the design bases were satisfied for control room habitability were not adequate in that the means for maintaining viable self-contained breathing apparatus capability for each person in the control room were not translated into procedures. In addition, the Licensee failed to translate the design bases for the Unit 1 standby gas treatment system (SGTS) into design specifications, and failed to perform comprehensive pre-operational testing of the SGTS to ensure that it met its design specifications.

At Millstone Unit 2, the Licensee failed to adequately update the surveillance requirements to reflect modifications to contact positions in the anticipated transient without scram (ATWS) mitigating system actuating circuitry. Also at Unit 2, the procedure requirements for the time of initiation of hydrogen monitoring following a LOCA were not consistent with the licensing and design bases.

In addition, there were a number of instances where the original design basis was inadequate or the original installation was incorrect. For example, at Units 2 and 3, the Licensee failed to remove plastic shipping plugs from Rosemount transmitters prior to installation, notwithstanding the vendor's instructions which required those plugs' replacement with stainless steel plugs. At Unit 2, the NRC found that nuclear instrumentation and post-LOCA hydrogen monitors were not single-failure proof.

At Millstone Unit 2, the Licensee's inspection of the containment sump screen mesh revealed that debris larger than the size specified in the design basis could pass through with potential adverse consequences to the operability of the emergency core cooling systems. The NRC also identified that the post-accident containment hydrogen monitor design at Millstone Unit 2 was flawed in that insufficient sample flow would be available at low containment pressures when the monitor must be operable.

Also at Unit 2, when it was found that postulated failures of the non-vital intake structure ventilation systems could cause the intake structure ambient temperature to exceed the design basis, the Licensee did not perform appropriate evaluations relative to the design basis before concluding that no modifications to equipment or the design basis were needed.

3. Inadequate Engineering and Modifications

The NRC identified a number of instances in which a modification was not installed in accordance with the design, a modification was inadequate, or a modification was based on incorrect design assumptions. In one example at Millstone Unit 1, the Licensee failed to maintain the design bases for the loss of normal power (LNP) logic. Specifically, a modification resulted in a single failure vulnerability of the LNP logic that would have prevented both emergency power sources from properly starting and sequencing the required loads. The Licensee also revised the Unit 1 maximum spent fuel pool temperature through an amendment to the Technical Specifications but failed to evaluate the impact of the change on the SGTS.

At Millstone Unit 2, both trains of service water were rendered inoperable when the strainer backwash line froze due to an undocumented modification that extended the backwash line through an opening under the wall to a point just outside the intake structure.

Also at Millstone Unit 2, the NRC identified that both trains of the post-accident sampling system have been inoperable since the steam generator replacement modification because higher containment pressures would have delayed taking a containment sample for 24 hours.

At Millstone Unit 3, the Licensee prepared a modification package for the high pressure safety injection thermal relief valves which relied on incorrect design assumptions because a previous modification had revised the design. In addition, the Licensee had no approved calculation to demonstrate the adequacy of the station blackout diesel generator battery at Millstone Unit 3.

Although the Licensee's own programs, such as the CMP, are intended to correct existing and prevent future deficiencies at the facilities, I have concluded that these programs by themselves are not sufficient, given the Licensee's history of poor performance in ensuring complete implementation of corrective action for both known degraded and non-conforming conditions and past violations of NRC requirements. In addition, the magnitude and scope of the design and configuration deficiencies currently being identified indicate multiple significant failures to comply with NRC regulations (e.g., 50.59, 50.71(e), etc.) The Licensee's history of poor performance, coupled with the magnitude and scope of its failure to maintain and control conformance of Millstone Units 1, 2, and 3 to their design bases, require resolution prior to plant restarts.

The extent and duration of the deficiencies identified also indicate ineffective implementation of the Licensee's oversight programs, including the NRC-approved quality assurance (QA) program. Effective oversight activities should have identified and led to corrective measures for design control deficiencies. One conclusion of ACR 7007 was that the Licensee's oversight organizations (Review Boards, Quality Assessment Section (QAS), Independent Safety Engineering Group, and Operating Experience) did not identify the pattern of Millstone Unit 1 UFSAR discrepancies to management; nor did they identify the significance of the pattern, or the effectiveness of corrective actions to prevent recurrence. In a July 2, 1996 letter to the NRC, the Licensee provided the preliminary findings of an independent Root Cause Evaluation Team chartered to determine the causes for these oversight failures. The team found that there was no history of escalating issues effectively and that QAS operated in an environment that did not lend itself to resolution of QAS-identified problems. Such findings of program weaknesses that represent poor oversight functions are not recent. It is apparent that the Licensee was aware of significant weaknesses in its oversight functions as early as 1991 and took no effective actions to correct those weaknesses. The Licensee's Performance Task Group Final Report, issued in September 1991, and Procedure Compliance Task Force Final Report, issued in October 1991, identified significant programmatic weaknesses affecting configuration management that either went unnoticed or were not corrected by the Licensee oversight functions.

It is necessary to ensure that the Licensee's programs to correct design control failures at Millstone Units 1, 2 and 3 are effective and that identification of degraded and non-conforming conditions and implementation of corrective actions are satisfactory and can effectively preclude repetition of these failures. For this reason, the NRC requires an independent verification of the adequacy of the results of the programs currently being implemented by the Licensee which are directed at resolving existing design and configuration management deficiencies. Accordingly, the Commission in this Order directs the Licensee to obtain the services of an organization, independent of the Licensee and its design contractors, to conduct a multi-disciplinary review of Millstone Units 1, 2, and 3. The review is to provide independent verification that, for the selected systems, the Licensee's CMP has identified and resolved existing problems, documented and utilized licensing and design bases, and established programs, processes and procedures for effective configuration management in the future. This review must be comprehensive, incorporating appropriate engineering disciplines, such that the NRC can be confident that the Licensee has been thorough in identification and resolution of problems.

III

On August 12, 1996, a transcribed meeting was conducted between the Licensee and the NRC staff regarding this matter. In response to the staff's concerns, the Licensee subsequently submitted a letter dated August 13, 1996, in which it agreed and committed to take a number of actions with respect to Millstone Units 1, 2, and 3. Specifically, the Licensee committed to have an independent team conduct an Independent Corrective Action Verification Program (ICAVP) at Millstone Units 1, 2, and 3. The Licensee committed that the corrective action verification program will include: (1) Conduct of an in-depth review of selected systems which will address control of the design and design basis since issuance of the operating license for each unit; (2) selection of systems for review based on risk/safety based criteria similar to those used in implementing the Maintenance Rule (10 CFR 50.65); (3) development and documentation of an audit plan that will provide assurance that the quality of results of the Licensee's problem identification and corrective action programs on the selected systems is representative of and consistent with that of other systems; (4) procedures and schedules for parallel reporting of findings and recommendations by the ICAVP team to both the NRC and the Licensee; and (5) procedures for the ICAVP team to comment on the Licensee's proposed resolution of the findings and recommendations. The Licensee also committed to the scope of the ICAVP review, encompassing modifications to the selected systems since initial licensing, including: (1) A review of engineering design and configuration control processes; (2) verification of current, as-modified plant conditions against design basis and licensing basis documentation; (3) verification that design and licensing bases requirements are translated into operating procedures, and maintenance and test procedures; (4) verification of system performance through review of specific test records and/or observation of selected testing of particular systems; and (5) review of proposed and implemented corrective actions for Licensee-identified design deficiencies.

I find that the Licensee's agreements and commitments as set forth in its letter of August 13, 1996 are acceptable and necessary.

In view of the foregoing, I have determined that public health and safety require that the Licensee's agreements and commitments in its August 13, 1996 letter be confirmed by this Order. The Licensee has agreed to this action. Pursuant to 10 CFR 2.202, I have also determined, based on the significance of the matters described above, as well as on the Licensee's consent, that the public health and safety require that this Order be immediately effective.

IV

Accordingly, pursuant to Sections 103, 104, 161b, 161i, 161o, 182 and 186 of the Atomic Energy Act of 1954, as amended, and the Commission's regulations in 10 CFR 2.202 and 10 CFR Part 50, *It is hereby ordered, effective immediately*, That:

1. The Licensee shall implement an Independent Corrective Action Verification Program (ICAVP) for each Millstone Unit to confirm that the plant's physical and functional characteristics are in conformance with its licensing and design bases. The ICAVP review shall begin after the Licensee has completed the problem identification phase of the CMP, including the activities of the QA organization. The ICAVP shall be performed and completed for each Unit, to the satisfaction of the NRC, prior to the Unit's restart.

2. The ICAVP is to be conducted by an independent verification team whose selection must be approved by the NRC. The ICAVP team shall provide input on its findings on an ongoing basis concurrently to both the Licensee and the NRC. The ICAVP team shall also periodically provide to the NRC its comments on the Licensee's proposed resolution of the team's findings and recommendations.

3. The ICAVP team shall provide for NRC review and approval, prior to implementation, a plan for the conduct of the team's review. The plan must describe (a) the conduct of an in-depth review of selected systems' design and design bases since issuance of the facilities' operating licenses; (b) risk/safety based criteria for selection of systems for review; (c) a description of the audit plan to provide assurance that the quality of results of the Licensee's problem identification and corrective action programs on the selected systems is representative of and consistent with that of other systems; (d) procedures and schedules for parallel reporting of findings of the ICAVP team to both the NRC and the Licensee; and (e) procedures for the ICAVP team to comment on the Licensee's proposed resolution of the team's findings and recommendations. The scope of the ICAVP effort shall encompass all modifications made to the selected systems since initial licensing, and shall include: (1) Review of engineering design and configuration control processes, (2) verification of current, as-modified conditions against design and licensing basis documentation, (3) verification that the design and licensing bases requirements have been translated into operating procedures, and maintenance and test procedures, (4) verification of system performance through review of specific test records and/or observation of selected testing, and (5) review of proposed and

implemented corrective actions for licensee-identified design deficiencies.

4. The Licensee shall provide written replies to the Regional Administrator, Region I and the Director, Office of Nuclear Reactor Regulation, addressing ICAVP team findings and recommendations discussed in reports made pursuant to item 3(d) above. The Licensee's written replies to ICAVP team findings and recommendations shall include a statement of agreement or disagreement with reasons for each ICAVP finding or recommendation, and of the status of implementation of corrective actions. Subsequent written replies shall be made until all corrective actions are implemented.

The Director, Office of Nuclear Reactor Regulation, may, in writing, relax or rescind this order upon demonstration by the Licensee of good cause.

V

The Licensee has, as described above, consented to the issuance of this Order and waived its right to request a hearing. Thus, any person adversely affected by this Order, other than the Licensee, may request a hearing within 20 days of its issuance. Where good cause is shown, consideration will be given to extending the time to request a hearing. A request for extension of time must be made in writing to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and include a statement of good cause for the extension. Any request for a hearing shall be submitted to the Secretary, U.S. Nuclear Regulatory Commission, ATTN: Chief, Docketing and Service Section, Washington, DC 20555. Copies of the hearing request shall also be sent to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555, to the Assistant General Counsel for Hearings and Enforcement at the same address, to the Regional Administrator, NRC Region I, 475 Allendale Road, King of Prussia, PA 19406–1415, and to the Licensee. If such a person requests a hearing, that person shall set forth with particularity the manner in which his interest is adversely affected by this Order and shall address the criteria set forth in 10 CFR 2.714(d).

If a hearing is requested by a person whose interest is adversely affected, the Commission will issue an Order designating the time and place of any hearings. If a hearing is held, the issue to be considered at such hearing shall be whether this Confirmatory Order should be sustained.

In the absence of any request for hearing, or written approval of an extension of time in which to request a hearing, the provisions specified in Section IV above shall be final 20 days from the date of this Order without further order or proceedings. If an extension of time for requesting a hearing has been approved, the provisions specified in Section IV shall be final when the extension expires if a hearing request has not been received. AN ANSWER OR A REQUEST FOR HEARING SHALL NOT STAY THE IMMEDIATE EFFECTIVENESS OF THIS ORDER.

Dated at Rockville, Maryland, this 14th day of August, 1996.

For the Nuclear Regulatory Commission.

William T. Russell,

Director, Office of Nuclear Reactor Regulation.

[FR Doc. 96–21162 Filed 8–19–96; 8:45 am]

BILLING CODE 7590–01–P

Training Requirements for Agreement State Personnel; Working Group

AGENCY: Nuclear Regulatory Commission.

ACTION: Establishment of Working Group on Training for Materials Licensing and Inspection.

SUMMARY: A working group consisting of representatives from Agreement States and from the Nuclear Regulatory Commission (NRC) has been formed to evaluate the ongoing evolution of training programs for Agreement State personnel, the criteria for evaluation of Agreement State programs in the area of training qualification, and the possible training options for Agreement State personnel.

FOR FURTHER INFORMATION CONTACT: Dennis M. Sollenberger, Office of State Programs (OSP), U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Telephone: 301–415–2819.

SUPPLEMENTARY INFORMATION: By letter dated November 14, 1995, Mr. Richard Ratliff, Chair, Organization of Agreement States (OAS), presented OAS concerns to the NRC including concerns in the area of training and requested that an operational committee or working group be established to consider identification of core courses, identification of additional training requirements for Agreement State personnel, and identification of acceptable alternate training options. The NRC responded to the letter on December 28, 1995, agreeing to the proposal to establish a working group to address the training issues of the OAS.

Over the last several years the training program conducted by NRC for Agreement State personnel has gone through an evolution in which the training developed and conducted for Agreement States has been merged with the training program for NRC staff. The overall coordination of this combined program is the responsibility of the Technical Training Division (TTD), Office for Analysis and Evaluation of Operational Data (AEOD). Other NRC offices and Regions provide input to the course content and training needs. The Office of State Programs (OSP) has collected and provided input on the Agreement State training needs.

The NRC has recently revised its training requirements for materials licensing and inspection staff. The requirements are now in one document, Inspection Manual Chapter 1246. The NRC has proposed that the Agreement State staff meet similar training requirements and that the Agreement State radiation control program directors formally establish staff qualification criteria and document that staff are qualified to independently perform work as they complete various training levels. The qualifications and training of Agreement State personnel have also been identified as one of the common performance indicators under the Integrated Materials Performance Evaluation Program (IMPEP) for evaluating Agreement State and NRC Regional materials regulatory programs. Specific criteria to benchmark this evaluation are needed to ensure uniformity for this program. This proposal was presented at the October 1995 All Agreement States meeting, which resulted in the above referenced letter from the OAS.

The Commission will discontinue the funding for Agreement State staff travel and contractor costs associated with Agreement State staff training beginning in fiscal year 1997. This action has prompted Agreement States to investigate alternate training methods to those made available by the NRC. The working group will not address the funding issue but will address possible alternate training methods.

Scope of Work

The NRC/OAS Training Working Group will address the Agreement State training issues as identified in the OAS letter of November 14, 1995 and other issues identified to the group by OAS or the NRC.

Tasks

In evaluating the potential training necessary for Agreement State personnel to have equivalent qualifications as NRC