

officials, foundation directors or other individuals that would be involved in funding or operating the model program. The principal decision makers for these entities are the desired participants.

The seminars will provide participants with the opportunity to tour CET skill training centers and to directly observe the CET job training model in action. It will provide the opportunity to meet and interact with CET instructors and students. Distinguished researchers will present study results related to their research of the CET program design. Information will be provided about CET's technical assistance program, the process for applying for technical assistance and the requirements for selection of CET model development sites. Applicants are encouraged to attend an information seminar prior to submitting their applications.

Applications

Applications will be accepted on a first come first serve basis starting August 1, 1995 from eligible applicants that meet the standards below.

To be considered, all applicants must meet the following standards:

(1) Applicants must be or become a JTPA service provider. If the service provider is not the JTPA administrative entity, the application must be submitted jointly with the JTPA administrative entity. A joint application should elaborate on the relationship between the service provider and the administrative entity. This should include descriptions of previous services; JTPA, JOBS and other training contract funding levels; and the level of cooperation between the service provider and the administrative entity.

(2) The proposed service provider must include a written commitment from their organization's board of directors.

(3) Applicants must be able to show that there is solid JTPA or other funding sources available to the proposed service provider. The applicants must commit to or have a reasonable likelihood of receiving operating funds of at least \$1,000,000 a year. A smaller amount may be acceptable for SDAs and other organizations operating in communities with limited funding resources. Any local procurement procedures that would be required prior to initiating the project, including those for a sole source award, if applicable, should be detailed. A timeline for the procurement process should be provided.

(4) The applicants must show a substantial startup funding

commitment. A minimum of \$250,000 is expected although lesser amounts with an explanation of why that level of funding is sufficient may be acceptable.

(5) The applicants must commit to sending staff for training throughout the technical assistance period.

(6) The applicants must commit to follow all key aspects of the CET-model training design, as discussed at the information seminar, including open admission of hard-to-serve client groups.

(7) The applicant must indicate its willingness to: (a) begin staff training within 3 months of notice of technical assistance award and, (b) open a new training site within 6 months of notice of technical assistance award. Those applicants who wish to schedule technical assistance after this period are encouraged to submit applications. Services to those applicants, however, will be contingent upon additional federal funding and may not be selected as part of this first come first serve request for proposals.

(8) The applicants must stipulate which of the following replication site service provider option it wishes to pursue:

- a. An incorporated SDA with 501(c)(3) status operates the program;
- b. A non-SDA, nonprofit, community-based or local education organization operates the program; or
- c. The Center for Employment Training operates the program.

(9) The applicants must state that they are willing to participate in evaluation studies.

Applications will be reviewed against the above standards. All acceptable applications will be ranked based upon date received. The initial 8 sites will be provided with technical assistance under this agreement. Additional service providers, including those applicants who request assistance outside the current time frame, will be served contingent upon the availability of funds. Applications which do not meet the standard will be notified with an explanation.

Signed at Washington, DC, this 30th day of June 1995.

James M. Aaron,

Director, Office of Employment and Training Programs.

[FR Doc. 95-16831 Filed 7-7-95; 8:45 am]

BILLING CODE 4510-30-M

NATIONAL AERONAUTICS AND SPACE ADMINISTRATION

[Notice (95-052)]

NASA Advisory Council (NAC), Space Science Advisory Committee (SScAC), Meeting

AGENCY: National Aeronautics and Space Administration.

ACTION: Notice of meeting.

SUMMARY: In accordance with the Federal Advisory Committee Act, Pub. L. 92-463, as amended, the National Aeronautics and Space Administration announces a forthcoming meeting of the NASA Advisory Council, Space Science Advisory Committee.

DATES: Monday, July 31, 1995, 8:30 a.m. to 5:00 p.m.; Tuesday, August 1, 1995, 8:30 a.m. to 5:00 p.m.

ADDRESSES: NASA Headquarters, Conference Room MIC 6-A&B-West, 300 E Street SW., Washington, DC 20546.

FOR FURTHER INFORMATION CONTACT: Kathy Dakon, Code SZ, National Aeronautics and Space Administration, Washington, DC 20546, 202/358-0732.

SUPPLEMENTARY INFORMATION: The meeting will be open to the public up to the capacity of the room. The agenda for the meeting is as follows:

- Status of Prior SScAC Recommendations
- Agency Streamlining
- Science Policy Guide
- FY 96 Budget Update
- Subcommittee Business

It is imperative that the meeting be held on these dates to accommodate the scheduling priorities of the key participants. Visitors will be requested to sign a visitor's register.

Dated: June 30, 1995.

Timothy M. Sullivan,

Advisory Committee Management Officer.

[FR Doc. 95-16833 Filed 7-7-95; 8:45 am]

BILLING CODE 7510-01-M

NUCLEAR REGULATORY COMMISSION

Report to Congress on Abnormal Occurrences October-December, 1994; Dissemination of Information

Section 208 of the Energy Reorganization Act of 1974, as amended, requires NRC to disseminate information on abnormal occurrences (AOs) (i.e., unscheduled incidents or events that the Commission determines are significant from the standpoint of public health and safety). During the

fourth quarter of CY 1994, the following incidents at NRC licensed facilities were determined to be AOs and are described below, together with the remedial actions taken. The events are also being included in NUREG-0090, Vol. 17, No. 4, ("Report to Congress on Abnormal Occurrences: October-December 1994"). This report will be available at NRC's Public Document Room, 2120 L Street NW. (Lower Level), Washington, DC 20037 about three weeks after the publication date of this **Federal Register** Notice.

Nuclear Power Plants

94-20 Core Shroud Cracking in Boiling Water Reactors

One of the AO reporting guidelines notes that a major deficiency in design, construction, or operation having safety implications requiring immediate attention can be considered an AO. A second reporting guideline notes that recurring incidents and incidents with implications for similar facilities (generic incidents) that create a major safety concern can be considered an AO.

Date and Place—From October 1993 through the present, various General Electric-designed boiling water reactors.

Nature and Probable Consequences—Intergranular stress corrosion cracking (IGSCC) of General Electric (GE)-designed boiling water reactor (BWR) reactor vessel internals has been identified as a technical issue of concern by both NRC and the industry. Core shroud cracking as a result of IGSCC was initially discovered overseas and later identified in operating BWR plants within the United States. Although no adverse consequences are expected at currently observed levels of shroud cracking, it has been postulated that a 360-degree through-wall core shroud crack in concert with a loss-of-coolant accident has the potential to lead to core damage.

NRC has been meeting every year since 1988 with the BWR Owners Group (BWROG) and GE to review the generic safety implications of potential failure of reactor internals, with IGSCC as one of the failure mechanisms of concern.

Cracking of BWR core shrouds was first observed in an overseas BWR in 1990. It was located in the heat affected zone of a circumferential weld in the beltline elevation of the shroud, and was reported by GE via Rapid Information Communication Services Information Letter (RICSIL) 054. The core shroud is a stainless steel cylinder which performs the following functions: (1) Separates feedwater in the reactor vessel's downcomer annulus from cooling water flowing through the

reactor core, (2) maintains core geometry, and (3) provides a refloodable volume under postulated accident conditions. The potential loss of a refloodable volume under accident conditions has the potential of resulting in core damage making BWR core shroud cracking the most significant concern related to potential failures of reactor internals reported during 1993 and 1994.

In response to this concern, several actions were taken by NRC. In a meeting with the BWROG in January 1992, the staff emphasized that a comprehensive program should be developed to address internals cracking and that the utilities should adopt an enhanced inspection program. In September 1993, Information Notice (IN) 93-79, "Cracking at the Beltline Region Welds in Boiling Water Reactors," was issued in response to the discovery of significant circumferential cracking of the core shroud welds at Brunswick Unit 1. (This event was also included in NRC's "Report to Congress on Abnormal Occurrences, October-December 1993." [NUREG-0090, Vol. 16, No. 4]).

Following the additional discovery of significant core shroud cracks at Dresden Unit 3 and Quad Cities Unit 1 in May and June 1994, respectively, NRC issued IN 94-42 "Cracking in the Lower Region of the Core Shroud in Boiling Water Reactors," June 7, 1994; IN 94-42 Supplement 1, July 19, 1994; and Generic Letter (GL) 94-03, "Intergranular Stress Corrosion Cracking of BWR Core Shrouds," July 25, 1994.

GL 94-03 requested that BWR licensees inspect their core shrouds at the next refueling outage, and perform a safety analysis to support continued operation of their facilities until corrective actions were implemented. During the same period of time, the BWROG initiated the BWR Vessels and Internals Project (BWRVIP) to facilitate industry response to the core shroud and internals cracking issues. Licensee responses to GL 94-03 were received during August and September 1994, and several BWR licensees began outages in September 1994.

Cause or Causes—IGSCC of BWR vessel internals is a time dependent material degradation process which is accelerated by the presence of crevices, residual stresses, material sensitization, irradiation, cold work and corrosive environments.

Actions Taken To Prevent Recurrence

Licensees—Several domestic BWR licensees performed visual examinations of their core shrouds in accordance with the recommendations of GE RICSIL 054 or GE Services

Information Letter (SIL) 572, which was issued in late 1993 and incorporates domestic experience.

NRC—Because of the extent of cracking observed, NRC evaluated safety concerns associated with the possibility of a 360-degree circumferential separation of the shroud following a pipe break. Such separation might either prevent full insertion of the control rods, or open a gap in the shroud large enough so that the resulting leakage would limit adequate core cooling by the emergency core cooling system. The accident scenarios of primary concern are the main steam line break and the recirculation line break, which are normally referred to as loss-of-coolant accidents.

The most serious event associated with cracks in the upper shroud welds is the steam line break, since the lifting forces generated may be sufficient to elevate the top guide and potentially affect the ability to insert rods. The most serious event associated with cracks in the lower elevations of the core shroud is the recirculation line break. A recirculation line break concurrent with a 360-degree through-wall weld failure could cause a lateral displacement of the shroud or opening of a crack, which would allow enough leakage through the shroud and out of the break affecting the ability to adequately cool the core.

NRC performed a probabilistic risk assessment of the consequences of shroud separation at the lower elevation for Dresden Unit 3 and Quad Cities Unit 1. The assessment estimated the potential contribution to core damage frequency from a cracked shroud. Assuming that severe shroud cracking (360-degree through-wall cracking) did exist, a large rupture of either a steam or recirculation line would have to occur to generate sufficiently large loads to move the shroud. No events involving a large rupture of a steam line or recirculation line have ever occurred, and probabilistic risk assessments have shown that such ruptures have a low probability of occurring. Furthermore, for welds in the upper portion of the shroud, such extensive degradation in and of itself can be detected during normal operation by a power/flow mismatch condition.

From the above evaluations, NRC made conservative estimates of the risk contribution to core damage from shroud cracking and concluded that immediate corrective actions are not necessary. Although immediate plant shutdowns to implement corrective actions are not necessary, degradation of the core shroud does have the potential to impact plant safety. The core shroud provides the important functions of

properly directing coolant flow through the core, maintaining core geometry, and providing a refloodable volume under postulated accident conditions. NRC therefore considers that 360-degree cracking of the shroud is a safety concern for the long term based on: (1) The potential to exceed the American Society of Mechanical Engineer Code's structural margins, if the cracks are sufficiently deep and continue to propagate through the subsequent operating cycle; and (2) the potential effects on the ability to protect against core damage.

Even though licensees have justified (through engineering evaluations) continued operation with significant cracks existing in core shrouds, BWRs with core shroud materials susceptible to IGSCC will eventually have to be repaired or modified to inhibit cracking and thereby assure structural integrity of the shrouds in the long term.

Due to the location and the extent of the cracking recently found, NRC and the BWROG agreed that additional attention to this issue was warranted. BWROG met with NRC on June 28, 1994, to announce the formation of BWRVIP, which is headed by several high level utility executives to direct its efforts. BWRVIP has since submitted documents which addressed an integrated safety assessment of the issue, inspection plans for the reactor internals, and generic criteria for repairs and flaw acceptance.

NRC has reviewed these documents and concurs with the BWRVIP recommended generic repair criteria and flaw assessment methodology. Inspection scope and methodology are still under consideration.

In addition to the above actions, in order to verify compliance with the structural integrity requirements of 10 CFR 50.55a and to assure that the risk associated with core shroud cracking remains low, NRC concluded that it is appropriate for BWR licensees to implement timely inspections and/or repairs, as appropriate, at their plants. To implement this position, NRC issued GL 94-03 (July 25, 1994) which requested BWR licensees to inspect their core shrouds by the next outage and to justify continued safe operation until all appropriate corrective actions have been implemented.

* * * * *

Other NRC Licensees

(Industrial Radiographers, Medical Institutions, Industrial Users, etc.)

94-21 Recurring Incidents of Administering Higher Doses Than Procedurally Allowed for Diagnostic Imaging at Ball Memorial Hospital in Muncie, Indiana

One of the AO reporting guidelines notes that a serious deficiency in management or procedural controls in a major area can be considered an AO.

Date and Place—October 1988 through June 1993; Ball Memorial Hospital; Muncie, Indiana.

Nature and Probable Consequences—On July 19, 1993, NRC was notified that nuclear medicine technologists employed by the licensee had increased the dosages of radiopharmaceuticals used in diagnostic studies. NRC was also informed that the technologists had falsified the required records of the dosages administered.

On July 21 through August 9, 1993, NRC conducted an inspection of the licensed facility. The inspection revealed that since 1988, nuclear medicine technologists employed by the licensee have been administering radiopharmaceutical dosages above the approved dose ranges for diagnostic image studies by as much as 40 percent. The inspection also verified that subsequent to administering high doses, the technologists entered false information in NRC-required records. The doses were increased for imaging studies of the lung, liver, bone, and gastrointestinal tract using technetium-99m and xenon-133.

NRC inspectors did not identify any medical misadministrations, as defined in 10 CFR 35.2, as a result of this practice of administering higher than approved doses for diagnostic imaging.

Cause or Causes—According to the licensee, one technologist told licensee officials that dosages were increased to minimize patient discomfort, to reduce imaging time for critically ill patients and to enhance the clarity of images for studies performed on obese patients.

Action Taken To Prevent Recurrence

Licensee—The licensee conducted an internal review. Based on the findings from this review, the licensee initially suspended two nuclear medicine technologists from all NRC-licensed activities. Subsequently, the licensee terminated one of the two individuals and the other individual was allowed to continue to perform duties that do not involve NRC-licensed activities.

The licensee also committed to a number of corrective actions. Some of the corrective actions include:

Assigning a pharmacist or a radiologist to verify all radioisotope dosages; implementing a unit dose system; obtaining the services of an assistant radiation safety officer; and conducting monthly and quarterly audits of the Nuclear Medicine Section for at least one year.

NRC—A special safety inspection was conducted by NRC from July 21 to August 9, 1993. Subsequent to that inspection, NRC conducted a followup review.

NRC issued a Confirmatory Action Letter on July 26, 1993, and a Confirmatory Order Modifying License on October 20, 1993. These documented specific procedures and verifications to prevent any further unauthorized increases in patient doses.

On May 23, 1994, NRC issued an Order against a former nuclear medicine technologist of the licensee. The Order required the following: (1) Prohibited the technologist from involvement in NRC-licensed activities for a period of one year; (2) required the technologist to provide a copy of the Order to any prospective employer who engages in NRC-licensed activities for a three-year period; and (3) required the technologist to notify NRC within 20 days of accepting employment involving NRC-licensed activities.

On May 27, 1994, the technologist requested a hearing and on September 26, 1994, a settlement agreement was reached. The settlement was reviewed and approved by the Atomic Safety and Licensing Board on October 3, 1994. The agreement resulted in the withdrawal of the requirement for the technologist to provide a copy of the Order to any prospective employer who engages in NRC-licensed activities. The settlement retained provisions (1) and (3) of the Order.

* * * * *

94-22 Medical Therapy Misadministration at Veterans Affairs Medical Center in Long Beach, California

One of the AO reporting guidelines notes that a therapeutic exposure to any part of the body not scheduled to receive radiation can be considered an AO.

Date and Place—August 9, 1994; Veterans Affairs Medical Center; Long Beach, California.

Nature and Probable Consequences—On August 9, 1994, the licensee's radiation safety officer (RSO) notified NRC of a misadministration involving a therapeutic dose of strontium-89 (Sr-89).

The RSO reported that a patient scheduled to receive 185 megabecquerel

(MBq) (5 millicurie [mCi]) of thallium-201 (a radiopharmaceutical not regulated by NRC) for a myocardial perfusion study was mistakenly administered 148 MBq (4 mCi) of Sr-89 (which is regulated by NRC). Based on the misadministration of the Sr-89, the licensee estimated that the patient received 250 centigray (250 rads) to the surface of the bone. The RSO reported that no action was taken to mitigate the consequences of the dose (i.e., administration of calcium as a blocking agent) because the patient had a preexisting heart condition which could have been exacerbated by administering calcium. The licensee also stated that medical experts were contacted to assist in an assessment of potential health effects to the patient. In addition, the licensee reported that with the exception of emergency procedures, it had voluntarily suspended all nuclear medicine procedures involving the intravenous administration of radiopharmaceuticals and had initiated an internal review of the misadministration.

On August 10, 1994, NRC issued a Confirmatory Action Letter to confirm the licensee's actions as stated above.

Cause or Causes—The cause of the misadministration was attributed to the administering technologist's failure to verify the isotope as well as the dosage (by reading the label on the syringe) prior to injection.

Actions Taken To Prevent Recurrence

Licensee—Corrective actions initially proposed by the licensee included the following: (1) Physically separating diagnostic unit dosages from therapeutic radiopharmaceutical dosages in the licensee's hot lab; (2) packaging unit dosages received from a local radiopharmacy in different containers, according to isotopes; and (3) retraining technologists in requirements for identifying radiopharmaceuticals prior to injection.

NRC—Two NRC inspectors conducted a special safety inspection on August 10-12 and 17-19, 1994, to review the circumstances associated with the misadministration and to review the licensee's corrective actions. In addition, NRC contracted a medical physician consultant to assist in its evaluation of the potential consequences of the patient's radiation exposure. The consultant stated that there were no adverse health effects to the patient.

An Enforcement Conference was held with the licensee on November 30, 1994, to discuss an apparent violation involving the failure of an individual working under the supervision of an

authorized user physician to follow the licensee's written radiation safety procedures. Additional concerns discussed during the conference included the licensee's use of an informal labeling system for unit radiopharmaceuticals which was identified as a potential programmatic weakness. The licensee presented information during the conference which supported its view that the error which led to the August 9, 1994, misadministration was an isolated failure rather than a programmatic problem.

Based on its review of information developed during the inspection and information provided during the Enforcement Conference, NRC concluded that the misadministration was the result of an isolated failure. A Notice of Violation was issued on December 29, 1994, for a violation involving the failure of an individual working under the supervision of a physician authorized user to follow the licensee's written procedures for verifying a radiopharmaceutical dose prior to administration to a patient. The violation was categorized as a Severity Level IV violation.

* * * * *

94-23 Medical Brachytherapy Misadministration at North Memorial Medical Center in Robbinsdale, Minnesota

One of the AO reporting guidelines notes that a therapeutic exposure to any part of a body not scheduled to receive radiation can be considered an AO.

Date and Place—August 3, 1994; North Memorial Medical Center; Robbinsdale, Minnesota.

Nature and Probable Consequences—On August 15, 1994, a licensee informed NRC that a patient received 1380 centigray (cGy) (1380 rads) to a wrong treatment site during a brachytherapy treatment for metastatic lung cancer.

On August 3, 1994, a catheter was inserted into the patient's bronchus and a ribbon containing 20 seeds of iridium-192 having a total activity of 673.4 megabecquerel (18.2 millicuries) was then inserted into the catheter and moved to the proper treatment location. The treatment plan was intended to deliver a prescribed dose of 2000 cGy (2000 rads) to the intended target. The treatment began at 11:15 a.m. on August 3, 1994, and continued until its scheduled completion at 10:15 a.m. on August 4, 1994.

At about 7 p.m. on August 3, 1994, a nurse informed the physician that the visible portion of the catheter appeared to be protruding approximately 25.4 to 30.5 centimeters (10 to 12 inches) from

the patient's nose. This was a significantly greater protrusion than previously observed, indicating that the catheter had moved from its initial placement. The nurse secured the catheter in place with additional tape. The physician stated that, based on the information available to him at that time, he determined that the catheter and ribbon had moved but that the tumor was receiving some radiation dose and therefore he continued the treatment. The iridium-192 seeds were removed on August 4 as planned. On August 4, 1994, a staff radiologist read the portable x-ray film taken on August 3, 1994, and indicated that the iridium implant was not seen.

Due to catheter displacement, the tumor dose was significantly reduced and estimated to be 620 cGy (620 rads) or 31 percent of the intended dose. The remaining dose of 1380 cGy (1380 rads) was delivered to an unintended site.

The patient was notified of the event by the treating physician on August 4, 1994, and again by another physician on August 17, 1994. The referring physician was informed by the treating physician on August 4, 1994.

An NRC medical consultant was retained to perform a clinical assessment of this misadministration. The medical consultant concluded that it is improbable that the patient will experience any long term consequences as a result of the exposure to the unintended treatment site.

Cause or Causes—The licensee has determined that the catheter movement caused a misadministration of the intended dose. Two possible explanations for the catheter movement could be the following: (1) Failure to properly secure the catheter in place with tape; or (2) nasal discharge decreasing the adhesive capability of the tape.

Action Taken To Prevent Recurrence

Licensee—The licensee's corrective actions include: amending the nursing staff procedure so that the attending physician will be contacted if there are further questions; directing nurses to follow the standing protocol for obtaining an administrative consult; providing additional inservice training; documenting the final length of the catheter in the patient chart; and documenting the catheter position on each visit to the patient's room.

NRC—NRC conducted a safety inspection from August 15 through September 7, 1994, to review the circumstances of the misadministration. One apparent violation and one area of concern were identified. An Enforcement Conference was held with

the licensee on October 11, 1994. Enforcement action is pending. NRC is continuing its review.

* * * * *

A copy of NUREG-0090, Vol. 17, No. 4 is available for inspection or copying for a fee at the NRC Public Document Room, 2120 L Street NW., (lower level), Washington, DC 20037, or at any of the nuclear power plant Local Public Document Rooms throughout the country.

Copies of this report (or any of the previous reports in this series), may be purchased from the Superintendent of Documents, U.S. Government Printing Office, Post Office Box 37082, Washington, DC 20013-7082. A year's subscription to the NUREG-0090 series publication, which consists of four issues, is also available.

Copies of the report may also be purchased from the National Technical Information Service, U.S. Department of Commerce, 5285 Port Royal Road, Springfield, VA 22161.

Dated at Rockville, MD this 3rd day of July 1995.

For the Nuclear Regulatory Commission.

John C. Hoyle,

Secretary of the Commission.

[FR Doc. 95-16808 Filed 7-7-95; 8:45 am]

BILLING CODE 7590-01-M

[Docket No. STN 50-456]

**Commonwealth Edison Company;
Braidwood Station, Unit 1;
Environmental Assessment and
Finding of No Significant Impact**

The U.S. Nuclear Regulatory Commission (the Commission) is considering issuance of an exemption from Facility Operating License No. NPF-72, issued to the Commonwealth Edison Company (the licensee), for Braidwood Station, Unit 1, located in Will County, Illinois.

Environmental Assessment

Identification of Proposed Action

The proposed action requests an exemption from certain requirements of 10 CFR 50.60, "Acceptance Criteria for Fracture Prevention Measures for Light-Water Nuclear Power Reactors for Normal Operation," to allow application of an alternate methodology to determine the low temperature overpressure protection (LTOP) setpoint for Braidwood Station, Unit 1. The proposed alternate methodology is consistent with guidelines developed by the American Society of Mechanical Engineers (ASME) Working Group on Operating Plant Criteria (WGOPC) to define pressure limits during LTOP

events that avoid certain unnecessary operational restrictions, provide adequate margins against failure of the reactor pressure vessel, and reduce the potential for unnecessary activation of pressure-relieving devices used for LTOP. These guidelines have been incorporated into Code Case N-514, "Low Temperature Overpressure Protection," which has been approved by the ASME Code Committee.

The content of this code case has been incorporated into Appendix G of Section XI of the ASME Code and published in the 1993 Addenda to Section XI. The NRC staff is revising 10 CFR 50.55a, which will endorse the 1993 Addenda and Appendix G of Section XI into the regulations.

The philosophy used to develop Code Case N-514 guidelines is to ensure that the LTOP limits are still below the pressure/temperature (P/T) limits for normal operation, but allow the pressure that may occur with activation of pressure-relieving devices to exceed the P/T limits, provided acceptable margins are maintained during these events. This philosophy protects the pressure vessel from LTOP events, and still maintains the Technical Specification P/T limits applicable for normal heatup and cooldown in accordance with Appendix G to 10 CFR Part 50 and Sections III and XI of the ASME Code. The exemption was requested by the licensee by letter dated November 30, 1994, and supplemented by letter dated May 11, 1995.

The Need for the Proposed Action

In 10 CFR 50.60 it states that all light-water nuclear power reactors must meet the fracture toughness and material surveillance program requirements for the reactor coolant pressure boundary as set forth in Appendices G and H to 10 CFR Part 50. Appendix G to 10 CFR 50 defines P/T limits during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime. It is specified in 10 CFR 50.60(b) that alternatives to the described requirements in Appendices G and H to 10 CFR Part 50 may be used when an exemption is granted by the Commission under 10 CFR 50.12.

To prevent transients that would produce pressure excursions exceeding the Appendix G P/T limits while the reactor is operating at low temperatures, the licensee installed an LTOP system. The LTOP system includes pressure relieving devices in the form of Power-Operated Relief Valves (PORVs) that are set at a pressure low enough that if a

transient occurred while the coolant temperature is below the LTOP enabling temperature, they would prevent the pressure in the reactor vessel from exceeding the Appendix G P/T limits. To prevent these valves from lifting as a result of normal operating pressure surges (e.g., reactor coolant pump starting, and shifting operating charging pumps) with the reactor coolant system in a water solid condition, the operating pressure must be maintained below the PORV setpoint.

In addition, in order to prevent cavitation of a reactor coolant pump, the operator must maintain a differential pressure across the reactor coolant pump seals. Hence, the licensee must operate the plant in a pressure window that is defined as the difference between the minimum required pressure to start a reactor coolant pump and the operating margin to prevent lifting of the PORVs due to normal operating pressure surges. The licensee's LTOP analysis indicates that using the Appendix G safety margins to determine the PORV setpoint would result in a pressure setpoint within its operating window, but there would be no margin for normal operating pressure surges. Therefore, operating with these limits could result in the lifting of the PORVs and cavitation of the reactor coolant pumps during normal operation. Therefore, the licensee proposed that in determining the PORV setpoint for LTOP events for Braidwood, the allowable pressure be determined using the safety margins developed in an alternate methodology in lieu of the safety margins required by Appendix G to 10 CFR Part 50. The alternate methodology is consistent with ASME Code Case N-514.

An exemption from 10 CFR 50.60 is required to use the alternate methodology for calculating the maximum allowable pressure for LTOP considerations.

Environmental Impacts of the Proposed Action

The Commission has completed its evaluation of the licensee's application.

Appendix G of the ASME Code requires that the P/T limits be calculated: (a) using a safety factor of two on the principal membrane (pressure) stresses, (b) assuming a flaw at the surface with a depth of one-quarter (1/4) of the vessel wall thickness and a length of six (6) times its depth, and (c) using a conservative fracture toughness curve that is based on the lower bound of static, dynamic, and crack arrest fracture toughness tests on material similar to the Braidwood reactor vessel material.