The SG tube voltage-based repair criteria presently in the Braidwood, Unit 1, TSs differ slightly from those proposed in the licensee's submittal dated August 15, 1995, in that the present repair criteria in the TSs were similar to those in the draft generic letter on the issue of ODSCC published by the staff on August 12, 1994, while the pending proposal is consistent with GL 95–05. This generic letter contains repair criteria slightly different from those contained in the earlier draft version. These differences reflect the staff's further review of this matter, including a review of comments by industry and the public.

In summary, the request for license amendments dated August 15, 1995, to adopt the voltage-based repair criteria in GL 95–05 will be considered by the staff only in the event that the pending request to raise the lower voltage limit from 1.0 volt to 3.0 volts can not be addressed in a timely manner.

While the voltage-based repair criteria for ODSCC flaws are applicable only to Braidwood, Unit 1, the pending request for license amendments involves both units in that the Braidwood Station has a set of TSs applicable to both units. Before issuance of the proposed license amendments, the Commission will have made findings required by the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations.

The Commission has made a proposed determination that the amendment request involves no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendments would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Consistent with Regulatory Guide (RG) 1.121, "Basis for Plugging Degraded PWR Steam Generator Tubes," Revision 0, August 1976, the traditional depth-based criteria for SG tube repair implicitly ensures that tubes accepted for continued service will retain adequate structural and leakage integrity during normal operating, transient, and postulated accident conditions. It is recognized that defects in tubes permitted to remain in service, especially cracks, occasionally grow entirely through-wall and develop small leaks. Limits on allowable primary-to-secondary leakage established in Technical Specifications ensure timely plant shutdown before the structural and leakage integrity of the affected tube is challenged.

The proposed license amendment request to implement voltage amplitude SG tube support plate APC for Braidwood Unit 1 meets the requirements of RG 1.121. The APC methodology demonstrates that tube leakage is acceptably low and tube burst is a highly improbable event during either normal operation or the most limiting accident condition, a postulated main steam line break (MSLB) event.

During transients, the tube support plate (TSP) is conservatively assumed to displace due to the thermal-hydraulic loads associated with the transient. This may partially expose a crack which is within the boundary of the TSP during normal operations to free span conditions. Burst is therefore conservatively evaluated assuming the crack is fully exposed to free span conditions. The structural eddy current bobbin coil voltage limit for free-span burst is 4.75 volts. This limit takes into consideration a 1.43 safety factor applied to the steam line break differential pressure that is consistent with RG 1.121 requirements. With additional considerations for growth rate assumptions and an upper 95% confidence estimate on voltage variability, the maximum voltage indication that could remain in service is given by the upper voltage repair limit equation in Generic Letter 95-05. For added conservatism, the allowable indication voltage is further reduced in the proposed amendment to a 1.0 volt confirmed ODSCC indication limit. All indications greater than 1.0 volt will be subject to an RPC examination. Tubes with RPC confirmed outside diameter stress corrosion cracking (ODSCC) indications will be plugged or sleeved. Any ODSCC indications between 1.0 volt and the upper voltage repair limit which are not confirmed as ODSCC will be allowed to remain in service since these indications are not as likely to affect tube structural integrity or leakage integrity over the next operating cycle as the indications that are detectable by both bobbin and rotating pancake coil (RPC) inspections.

The eddy current inspection process has been enhanced to address RG 1.83, "Inservice Inspection of PWR Steam Generator Tubes," Revision 1, July 1975, considerations as well as the EPRI SG Inspection Guidelines. Enhancements in accordance with Generic Letter 95–05 are in place to increase detection of ODSCC indications and to ensure reliable, consistent acquisition and analysis of data. Based on the conservative selection of the voltage criteria and the increased ability to identify ODSCC, the probability of tube failure during an accident is also not significantly increased due to application of requested APC.

Modification of the Braidwood Specifications for conformance with Generic Letter 95–05 requirements does not impact any accidents previously evaluated. The decrease in the allowed burst probability from $2.5 \times 10_{-2}$ to $1.0 \times 10_{-2}$ is conservative.

Calculations conducted for Braidwood have shown that the resulting 2-hour doses at the site boundaries will not currently exceed an appropriately small fraction of 10 CFR 100 dose guideline values in conjunction with the predicted MSLB leakage calculated in accordance with this submittal and a DE I-131 level of 1.0 µCi/gm. The site allowable leakage calculated using a DE I-131 level of 1.0 µCi/gm is 9.4 gallons per minute (gpm). This leakage includes accident leakage and the allowed 0.1 gpm primary-to-secondary leakage of the 3 unfaulted SGs per TS 3.4.6.2.c. However, in order to provide a defense in depth approach to application of this requested APC and to envelope any future increases in MSLB leakage due to tube degradation, Braidwood is lowering the RCS DE I-131 levels to 0.35 µCi/gm for all future cycles until SG replacement. The site allowable leak rate calculated using 0.35 µCi/gm DE I-131 is 26.8 gpm. This leakage also includes accident leakage and the allowed 0.1 gpm primary-tosecondary leakage of the 3 unfaulted SGs per TS 3.4.6.2.c. Lowering the limit to 0.35 μ Ci/ gm DE I-131 is conservative and will not increase the probability or consequences of any accidents previously evaluated.

Renewal of the 1.0 volt IPC for Braidwood Unit 1 does not adversely affect steam generator tube integrity and results in acceptable dose consequences. Therefore, the proposed license amendment request does not result in any significant increase in the probability or consequences of an accident previously evaluated within the Braidwood Updated Final Safety Analysis Report.

2. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Renewal of the proposed SG tube APC for Braidwood Unit 1 does not introduce any significant changes to the plant design basis. Use of the criteria does not provide a mechanism which could result in an accident outside the tube support plate elevations since industry experience indicates that ODSCC originating within the tube support plate does not extend significantly beyond the thickness of the support plate. This criteria only applies to ODSCC contained within the region of the tube bounded by the tube support plate. Therefore, neither a single or multiple tube rupture event would be expected in a steam generator in which APC has been applied.

In addressing the combined effects of Loss of Coolant Accident (LOCA) coincident with a Safe Shutdown Earthquake (SSE) on the SG (as required by General Design Criteria 2), it has been determined that tube collapse of select tubes may occur in the SGs at some plants, including Braidwood Unit 1. There are two issues associated with SG tube collapse. First, the collapse of SG tubing reduces the RCS flow area through the tubes. The reduction in flow area increases the resistance to flow of steam from the core during a LOCA which, in turn, may potentially increase Peak Clad Temperature (PCT). Second, there is a potential that partial through-wall cracks in tubes could progress to through-wall cracks during tube deformation or collapse. A number of tubes